September 29, 1983

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Docket No. 50-244
LS05-83-09-049

Mr. John E. Maier
Vice President
Electric and Steam Production
Rochester Gas & Electric Corporation
89 East Avenue
Rochester, New York 14649

Dear Mr. Maier:

SUBJECT: RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS - GINNA

The Commission has issued the enclosed Amendment No.57 to Provisional Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant (Ginna) in response to your submittals of August 12, 1982, January 10 and March 4, 1983, which supersede your submittals dated February 14 and May 29, 1979.

The amendment incorporates changes to the Technical Specifications developed for the purpose of keeping releases of radioactive material to unrestricted areas during normal operations, including expected operational occurrences, as low as reasonably achievable.

Technical Specification provisions approved by the amendment are to be implemented by January 1, 1984. This schedule allows for procedures generation and implementation, completion of the computer system upgrade and training of personnel necessary to fulfill the requirements of this amendment.

Based on our review, as supported by the Technical Evaluation performed by our contractor, Franklin Research Center, we conclude that the proposed Radiological Effluent Technical Specifications (RETS) meet the intent of the NRC staff's model RETS for pressurized water reactors, NUREG-0472, Revision 2, February 1, 1980. In addition, the Offsite Dose Calculation Manual (ODCM) used as a reference document, uses documented and approved methods that are consistent with the methodology and guidelines in NUREG-0133, and is, therefore, an acceptable reference.

During our review of your submittal we found it necessary to make minor modifications to your proposed Technical Specifications. We have discussed these modifications with your representative and have mutually agreed upon them.

S E01

DSU USE 07

A Notice of Consideration of Issuance of Amendment to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested action was published in the Federal Register on July 20, 1983 (48 FR 33088). No request for hearing and no public or State comments were received.

OFFICE						
SURNAME	8310070365 830929 PDR ADOCK 05000244 P PDR					
DATE						

NRC FORM 318 (10/80) NRCM 0240 **OFFICIAL RECORD COPY** ☆ U.S. GPO 1983—400-247

Copies of our related Safety Evaluation and its supporting Technical Evaluation Report (TER) prepared by our contractor Franklin Research Center are also enclosed, This action will appear in the Commission's Monthly Notice publication in the Federal Register.

Original signed by
Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosures:
1. Amendment No. 57 to
   License No. DPR-18
2. Safety Evaluation, including the TER

cc w/enclosures:
See next page

Check for Pattison Comments immediately before issuing & any comeback to OELD JB

Done 9/30/83 JEL/gc

OFFICE	DL:ORB #5	DL:ORB #5	DL:ORB #5	DL:ORB #5	OELD	DL:AD/SA	DL:ORB #4
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

September 28, 1983

Docket No. 50-244
LS05-83-09-049

Mr. John E. Maier
Vice President
Electric and Steam Production
Rochester Gas & Electric Corporation
89 East Avenue
Rochester, New York 14649

Dear Mr. Maier:

SUBJECT: RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS - GINNA

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During our review of your submittal we found it necessary to make minor modifications to your proposed Technical Specifications. We have discussed these modifications with your representative and have mutually agreed upon them.

A Notice of Consideration of Issuance of Amendment to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested action was published in the Federal Register on July 20, 1983 (48 FR 33088). No request for hearing and no public or State comments were received.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C(2) of Provisional Operating License No. DPR-18 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A as revised through Amendment No. 57 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance. However, the Technical Specification provisions are to be implemented by January 1, 1984.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: September 28, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 57

PROVISIONAL OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages* contain the captioned amendment number and marginal lines which indicate the area of changes.

Remove Pages	Insert Pages
i and ii	i and ii
iii	--
1-2	1-2
--	1-2a
1-5	1-5
--	1-6 and 1-7
3.5-1 and 3.5-2	3.5-1 through 3.5-2c
3.5-14	3.4-14 through 3.5-17
3.9-1 through 3.9-7	3.9-1 through 3.9-14
--	3.16-1 through 3.16-9
4.1-1	4.1-1 through 4.1-1a
--	4.1-2**
4.1-6	4.1-6
--	4.1-13 and 4.1-14
4.10-1 through 4.10-5	4.10-1 through 4.10-5
4.12-1 through 4.12-5	4.12-1 through 4.12-8
5.1-1	5.1-1 and 5.1-2
--	5.5-1 and 5.5-2

* The Technical Specification provisions approved by this action are to be implemented by January 1, 1984.

** This page is included for pagination purposes only; there are no changes to the provisions contained thereon.

Remove Pages	Insert Pages
6.5-10	6.5-10 and 6.4-10a
6.8-1	6.8-1 and 6.8-2
6.9-1 and 6.9-2	6.9-1, including 1a and 1b, through 6.9-3
6.9-7 through 6.9-10	6.9-7 thorugh 6.9-10
--	6.15-1
--	6.16-1
--	6.17-1 and 6.17-2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

ROCHESTER GAS AND ELECTRIC CORPORATION

DOCKET NO. 50-244

R. E. GINNA NUCLEAR POWER PLANT

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 57
License No. DPR-18

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Rochester Gas and Electric Corporation (the licensee) dated August 12, 1982, as supplemented January 10 and March 4, 1983 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public; and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public, and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

Copies of our related Safety Evaluation and its supporting Technical Evaluation Report (TER) prepared by our contractor Franklin Research Center are also enclosed, This action will appear in the Commission's Monthly Notice publication in the Federal Register.

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Enclosures:
1. Amendment No. 57 to
   License No. DPR-18
2. Safety Evaluation, including the TER

cc w/enclosures:
See next page

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TABLE OF CONTENTS (cont'd)

1.5 Operating

Performing all intended functions in the intended manner.

1.6 Degree of Redundancy (Instrument Channels)

The difference between the number of operable channels and the number of channels which, when tripped, will cause an automatic system trip.

1.7 Instrument Surveillance

1.7.1 Channel Calibration

The adjustment, as necessary, of the channel output so that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The Channel Calibration shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the Channel Functional Test. The Channel Calibration may be performed by any series of sequential, overlapping or total channel steps so that the entire channel is calibrated.

1.7.2 Channel Check

The qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrumentation channels measuring the same parameter.

1.7.3 Channel Functional Test

a. Analog channels - the injection of a simulated or source signal into the channel as close to the sensor as practicable to verify operability including alarm and/or trip functions.

b. Bistable channels - the injection of a simulated or source signal into the sensor to verify operability including alarm and/or trip function.

1.7.4 Source Check

The qualitative assessment of channel response when the channel sensor is exposed to a radioactive source.

1.12 **Frequency Notation**

The frequency notation specified for the performance of surveillance requirements shall correspond to the intervals defined below.

Notation	Frequency
S, Each Shift	At least once per 12 hours
D, Daily	At least once per 24 hours
Twice per week	At least once per 4 days and at least twice per 7 days
W, Weekly	At least once per 7 days
B/W, Biweekly	At least once per 14 days
M, Monthly	At least once per 31 days
B/M, Bimonthly	At least once per 62 days
Q, Quarterly	At least once per 92 days
SA, Semiannually	At least once per 6 months
A, Annually	At least once per 12 months
R	At least once per 18 months
S/U	Prior to each startup
N.A.	Not Applicable
P	Prior to each startup if not done previous week
PR	Within 12 hours prior to each release

1.13 **Offsite Dose Calculation Manual (ODCM)**

The ODCM is a manual containing the methodology and parameters to be used for calculating the offsite doses due to liquid and gaseous radiological effluents, in calculation of liquid and gaseous effluent monitoring instrumentation alarm/trip setpoints, and in the conduct of the environmental radiological monitoring program.

1.14 Process Control Program (PCP)

The PCP is a manual outlining the method for processing wet solid wastes and for solidification of liquid wastes. It shall include the process parameters and evaluation methods used to assure meeting the requirements of 10 CFR Part 71 prior to shipment of containers of radioactive waste from the site.

1.15 Solidification

Solidification shall be the conversion of radioactive wastes from liquid systems to a homogeneous solid.

1.16 Purge-Purging

Purge or purging is the controlled process of discharging air or gas from a confined space to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confined space.

1.17 Venting

Venting is the controlled process of discharging air or gas from a confined space to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air is not provided or required.

1.18 Dose Equivalent I-131

The dose equivalent I-131 shall be that concentration of I-131 which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The dose conversion factors used for this calculation shall be those for the adult thyroid dose via inhalation, contained in NRC Regulatory Guide 1.109 Rev. 1 October 1977.

3.5 **Instrumentation Systems**

Applicability:

Applies to plant instrumentation systems.

Objective:

To delineate the conditions of the plant instrumentation and safety circuits and to limit the release of radioactive materials.

Specification:

3.5.1 Operational Safety Instrumentation

3.5.1.1 The number of Minimum Operable Channels for instrumentation shown on Tables 3.5-1 through 3.5-3 shall be OPERABLE for plant operation at rated power.

3.5.1.2 In the event the number of channels of a particular sub-system in service falls below the limit given in the columns entitled Minimum Operable Channels, operation shall be limited according to the requirement shown in the last column of Tables 3.5-1 through 3.5-3.

3.5.2 Accident Monitoring Instrumentation

3.5.2.1 The accident monitoring instrumentation channels shown in Table 3.5-4 shall be operable whenever the reactor is at hot shutdown or is critical.

3.5.2.2 While critical, with the number of operable accident monitoring instrumentation channels less than the Total Number of Channels shown in Table 3.5-4, either restore the inoperable channel(s) to operable status within 7 days, or be in at least hot shutdown within the next 12 hours.

3.5.2.3 While critical, with the number of operable accident monitoring instrumentation channels less than the MINIMUM CHANNELS OPERABLE requirements of Table 3.5-4, either restore the inoperable channel(s) to operable status within 48 hours or be in at least hot shutdown within the next 12 hours.

3.5.3 Engineered Safety Feature Actuation Instrumentation

3.5.3.1 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels shown in Tables 3.5-2 and 3.5-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.5-5.

3.5.3.2 With an instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.5-5, declare the channel inoperable and apply the applicable ACTION requirement of Tables 3.5-2 and 3.5-3 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint Value.

3.5.3.3 With an instrumentation channel inoperable, take the action shown in Tables 3.5-2 and 3.5-3.

3.5.4 Radioactive Effluent Monitoring Instrumentation

3.5.4.1 The radioactive effluent monitoring instrumentation shown in Table 3.5-6 shall be operable at all times with alarm and/or trip setpoints set to ensure that the limits of Specifications 3.9.1.1 and 3.9.2.1 are not exceeded. Alarm and/or trip setpoints shall be established in accordance with calculational methods set forth in the Offsite Dose Calculation Manual.

3.5.4.2 If the setpoint for a radioactive effluent monitor alarm and/or trip is found to be higher than required, one of the following three measures shall be taken immediately:

(i) the setpoint shall be immediately corrected without declaring the channel inoperable; or

(ii) immediately suspend the release of effluents monitored by the affected channel; or

(iii) declare the channel inoperable.

3.5.4.3 If the number of channels which are operable is found to be less than required, take the action shown in Table 3.5-6.

Basis

During plant operations, the complete instrumentation systems will normally be in service. Reactor safety is provided by the Reactor Protection System, which automatically initiates appropriate action to prevent exceeding established limits. Safety is not compromised, however, by continuing operation with certain instrumentation channels out of service since provisions were made for this in the plant design. This specification outlines limiting conditions for operation necessary to preserve the effectiveness of the reactor control and protection system when any one or more of the channels is out of service.

Almost all reactor protection channels are supplied with sufficient redundancy to provide the capability for channel calibration and test at power. Exceptions are backup channels such as reactor coolant pump breakers. The removal of one trip channel is accomplished by placing that channel bistable in a tripped mode; e.g., a two-out-of-three circuit becomes a one-out-of-two circuit. Testing does not trip the system unless a trip condition exists in a concurrent channel.

The operability of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendations of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendation".

The radioactive liquid effluent instrumentation is provided to monitor and/or control, as applicable, the releases of radioactive materials in liquid effluents. The alarm and/or trip setpoints for these instruments are calculated in accordance with the ODCM to ensure that alarm and/or trip will occur prior to exceeding the limits of 10 CFR Part 20. The operability and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents. The alarm and/or trip setpoints for these instruments are calculated in accordance with the ODCM to ensure that alarm and/or trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The operability and use of this instrumentation is consistent with the requirements of General Design Criterion 64 of Appendix A to 10 CFR Part 50.

Reference

FSAR - Section 7.2.1

TABLE 3.5-5 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
7. LOSS OF VOLTAGE		
a. 480 V Safeguards Bus Under-voltage (Loss of Voltage)	see Figure 2.3-1	
b. 480 V Safeguards Bus Under-voltage (Degraded Voltage)	see Figure 2.3-1	
8. ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INTERLOCKS		
a. Pressurizer Pressure, (block, unblock SI)	$\leq$ 2000 psig	$\leq$ 2000 psig

Note 1: A positive 11% error has been included in the setpoint to account for errors which may be introduced into the steam generator level measurement system at a containment temperature of 286°F as determined by an evaluation performed on temperature effects on level systems as required by IE Bulletin 79-21.

Note 2: This setpoint is from inverse time curve for CVT relay (406C883) with tap setting of 82 volts and time dial setting of 1. Delay at 62% voltage is 3.6 seconds. The allowable values are ±5% of the trip setpoint.

Note 3: The trip setpoints for containment ventilation isolation while purging shall be established in accordance with calculational methods set forth in the ODCM.

*Allowable Values are those values assumed in accident analysis.

TABLE 3.5-6

Radioactive Effluent Monitoring Instrumentation

		Minimum Channels Operable	Action
1.	Gross Activity Monitors (Liquid)		
	a. Liquid Radwaste (R-18)	1	1
	b. Steam Generator Blowdown (R-19)	1*	2
	c. Turbine Building Floor Drains (R-21)	1	3
	d. High Conductivity Waste (R-22)	1	1
	e. Containment Fan Coolers (R-16)	1	3
	f. Spent Fuel Pool Heat Exchanger (R-20)	1	3
2.	Plant Ventilation		
	a. Noble Gas Activity (R-14) (Providing Alarm and Isolation of Gas Decay Tanks)	1	4
	b. Particulate Sampler (R-13)	1	5
	c. Iodine Sampler (R-10B or R-14A)	1	5
3.	Containment Purge Vent		
	a. Noble Gas Activity (R-12)	1+	(see Table 3.5-3 & note 2 thereto)
	b. Particulate Sampler (R-11)	1+	(see Table 3.5-3 & note 2 thereto)
	c. Iodine Sampler (R-10A or R-12A)	1+	5
4.	Air Ejector Monitor (R-15 or R-15A)	1**	6
5.	Waste Gas System Oxygen Monitor	1	7

*Not required when Steam Generator Blowdown is being recycled (i.e. not released)
+Required only during containment purges
**Not required during Cold or Refueling Shutdown

TABLE 3.5-6 (Continued)

Table Notation

Action 1 - If the number of operable channels is less than required by the Minimum Channels Operable requirement, effluent releases from the tank may continue for up to 14 days, provided that prior to initiating a release:

1. At least two independent samples of the tank's contents are analyzed, in accordance with Specification 4.12.1.1.a, and

2. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge line valving;

Otherwise, suspend release of radioactive effluents via this pathway.

Action 2 - When Steam Generator Blowdown is being released (not recycled) and the number of channels operable is less than required by the Minimum Channels Operable requirement, effluent releases via this pathway may continue for up to 31 days, provided grab samples are analyzed for gross radioactivity (beta or gamma) at a limit of detection of at most 10-7 uCi/gram:

1. At least once per 8 hours when the concentration of the secondary coolant is > 0.01 uCi/gram dose equivalent I-131.

2. At least once per 24 hours when the concentration of the secondary coolant is $\leq$ 0.01 uCi/gram dose equivalent I-131.

Action 3 - If the number of operable channels is less than required by the Minimum Channels operable requirement, effluent releases via this pathway may continue for up to 31 days provided that at least once per 24 hours grab samples are analyzed for gross radioactivity (beta or gamma) at a limit of detection of at most 10-7 uCi/gm.

Action 4 - If the number of operable channels is less than required by the Minimum Channels Operable requirement, effluent releases via this pathway may continue for up to 31 days provided grab samples are taken at least once per 8 hours and these samples are analyzed for isotopic activity within 24 hours.

TABLE 3.5-6 (Continued)

Table Notation

Action 5 - If the number of operable channels is less than required by the Minimum Channels Operable requirement, effluent releases via this pathway may continue for up to 31 days, provided samples are continuously collected as required by Table 4.12-2 Item E with auxiliary sampling equipment.

Action 6 - If the number of operable channels is less than required by the Minimum Channels Operable and the Secondary Activity is $\leq 1 \times 10^{-4}$ uCi/gm, effluent releases may continue via this pathway provided grab samples are analyzed for gross radioactivity (beta or gamma) at least once per 24 hours. If the secondary activity is greater than 1×10^{-4} uCi/gm, effluent releases via this pathway may continue for up to 31 days provided grab samples are taken every 8 hours and analyzed within 24 hours.

Action 7 - If the channel is out of service, a sample of the gas from each active gas decay tank shall be analyzed for oxygen content at least once every 4 hours.

3.9 **Plant Effluents**

Applicability

Applies to the controlled release of radioactive liquids and gases from the plant.

Objective

To define the conditions for release of radioactive liquid and gaseous wastes.

Specifications

3.9.1 **Liquid Effluents**

3.9.1.1 Concentration

3.9.1.1.a The release of radioactive liquid effluents shall be such that the concentration in the circulating water discharge does not exceed the limits specified in accordance with Appendix B, Table II, Column 2 and Notes thereto of 10CFR20. For dissolved or entrained noble gases the total activity due to dissolved or entrained noble gases shall not exceed 2 x 10^{-4} uCi/ml.

3.9.1.1.b If the concentration of radioactive material in the circulating water discharge exceeds the limits of 3.9.1.1.a, measures shall be initiated to restore the concentration to within those limits as soon as practicable.

3.9.1.2 Dose

3.9.1.2.a The dose or dose commitment to an individual as calculated in the ODCM from radioactive materials in liquid effluents released to unrestricted areas shall be limited:

(i) During any calendar quarter to $\leq$ 1.5 mrem to the total body and to $\leq$ 5 mrem to any organ, and

(ii) During any calendar year to $\leq$ 3 mrem to the total body and to $\leq$ 10 mrem to any organ.

3.9.1.2.b Whenever the calculated dose resulting from the release of radioactive materials in liquid effluents exceeds the quarterly limits of 3.9.1.2.a(i), a Special Report shall be submitted to the Commission within thirty days which includes the following information:

(i) Identification of the cause for exceeding the dose limit.

(ii) Corrective actions taken and/or to be taken to reduce the releases of radioactive material in liquid effluents to assure that subsequent releases will remain within the above limits.

(iii) The results of the radiological analyses of the nearest public drinking water source, and an evaluation of the radiological impact due to licensee releases on finished drinking water with regard to the requirements of 40 CFR 141 Safe Drinking Water Act.

3.9.1.3 Liquid Waste Treatment

3.9.1.3.a The liquid water treatment system shall be used to reduce the radioactive materials in liquid wastes prior to their discharge, if necessary, to assure that the cumulative dose due to liquid effluent releases when averaged over 31 days does not exceed 0.06 mrem to the total body or 0.2 mrem to any organ.

3.9.1.3.b If the liquid radwaste treatment system is not operable for more than 31 days and if radioactive liquid waste is being discharged without treatment resulting in doses in excess of Specification 3.9.1.3.a, a Special Report shall be submitted to the Commission within thirty days which includes the following information:

(i) Identification of equipment or subsystems not operable and the reasons.

(ii) Action(s) taken to restore the inoperable equipment to operable status.

(iii) Summary description of action(s) taken to prevent a recurrence.

3.9.2 <u>Gaseous Wastes</u>

3.9.2.1 Dose Rate

3.9.2.1.a The instantaneous dose rate, as calculated in the ODCM, due to radioactive materials released in gaseous effluents from the site shall be limited to the following values:

(i) The dose rate for noble gases shall be $\leq$ 500 mrem/yr to the total body and $\leq$ 3000 mrem/yr to the skin, and

(ii) The dose rate for all radioiodines, radioactive materials in particulate form, and radionuclides other than noble gases with half-lives greater than 8 days shall be $\leq$ 1500 mrem/yr to any organ.

3.9.2.1.b For unplanned release of gaseous wastes, compliance with 3.9.2.1.a may be determined by averaging over a 24-hour period.

3.9.2.1.c If the calculated dose rate of radioactive materials released in gaseous effluents from the site exceeds the limits of 3.9.2.1.a or 3.9.2.1.b, measures shall be initiated to restore releases to within those limits as soon as practicable.

3.9.2.1.d Compliance with 3.9.2.1.a and 3.9.2.1.b shall be determined by considering the applicable ventilation system flow rates. These flow rates shall be determined at the frequency required by Table 4.1-5.

3.9.2.2 Dose (10 CFR Part 50, Appendix I)

3.9.2.2.a The air dose, as calculated in the ODCM, due to noble gases released in gaseous effluents from the site shall be limited to the following:

(i) During any calendar quarter to $\leq$ 5 mrad for gamma radiation and to $\leq$ 10 mrad for beta radiation.

(ii) During any calendar year to $\leq$ 10 mrad for gamma radiation and to $\leq$ 20 mrad for beta radiation.

3.9.2.2.b The dose to an individual, as calculated in the ODCM, from radioiodine, radioactive materials in particulate form and radionuclides other than noble gases with half-lives greater than eight days released with gaseous effluents from the site shall be limited to the following:

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(i) During any calendar quarter to $\leq$ 7.5 mrem to any organ.

(ii) During any calendar year to $\leq$ 15 mrem to any organ.

3.9.2.2.c Whenever the calculated dose to an individual resulting from noble gases or from radionuclides other than noble gases exceeds the quarterly limits of 3.9.2.2.a(i) or 3.9.2.2.b(i) a Special Report shall be submitted to the Commission within thirty days which includes the following information:

(i) Identification of the cause for exceeding the dose limit.

(ii) Corrective actions taken and/or to be taken to reduce releases of radioactive material in gaseous effluents to assure that subsequent releases will be within the above limits.

3.9.2.3 Gaseous Waste Treatment

3.9.2.3.a The gaseous radwaste treatment system shall be used to reduce radioactive materials in gaseous waste prior to their discharge, if necessary, to assure that the cumulative air dose due to gaseous effluent releases to unrestricted areas when averaged over 31 days does not exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation to the maximally exposed individual.

3.9.2.3.b The appropriate portions of the ventilation exhaust system shall be used to reduce radioactive materials in gaseous waste prior to their discharge, if necessary, to assure that the cumulative dose due to gaseous effluent releases from the site when averaged over 31 days does not exceed 0.30 mrem to any organ.

3.9.2.3.c If the gaseous radwaste treatment system or ventilation exhaust system is inoperable for more than 31 days and if gaseous waste is being discharged without treatment resulting in doses in excess of Specifications 3.9.2.3.a or 3.9.2.3.b, a Special Report shall be submitted to the Commission within thirty days which includes the following information:

(i) Identification of equipment or subsystems not operable and the reasons.

(ii) Action(s) taken to restore the inoperable equipment to operable status.

(iii) Summary description of action(s) taken to prevent a recurrence.

3.9.2.4 Dose (40 CFR Part 190)

3.9.2.4.a If the calculated dose from the release of radioactive materials from the plant in liquid or gaseous effluents exceeds twice the limits of Specifications 3.9.1.2.a, 3.9.2.2.a, or 3.9.2.2.b, a Special Report shall be submitted to the Commission within thirty days and subsequent releases shall be limited so that the dose or dose commitment to a real individual is limited to $\leq$ 25 mrem to the total body or any organ (except thyroid, which is limited to $\leq$ 75 mrem) for the calendar year that includes the release(s) covered by this report.

This report shall include an analysis which demonstrates that radiation exposures to all real individuals from the plant are less than the 40 CFR Part 190 limits in accordance with methods set forth in the ODCM. Otherwise, the report shall request a variance from the Commission to permit releases to exceed 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

3.9.2.5 Explosive Gas Mixture

3.9.2.5.a The concentration of oxygen in each gas decay tank shall be limited to ≤ 2% by volume.

3.9.2.5.b If the concentration of oxygen in a gas decay tank is > 2% by volume but ≤ 4% by volume, restore the concentration of oxygen to within the limit within 48 hours.

3.9.2.5.c If the concentration of oxygen in a gas decay tank is > 4% by volume, immediately remove that tank from "reuse" or "in service" status and reduce the concentration of oxygen to ≤ 2% within 48 hours if such measures do not conflict with other radiological limits or procedures.

3.9.2.6 Waste Gas Decay Tanks

3.9.2.6.a The quantity of radioactivity contained in each waste gas decay tank shall be limited to less than or equal to 100,000 curies of noble gas (considered as Xe-133) at all times.

3.9.2.6.b If the quantity of radioactive material in any waste gas decay tank exceeds the limit of 3.9.2.6.a, immediately suspend all additions of radioactive material to the tank and reduce the tank contents within 48 hours if such measures do not conflict with other radiological limits or procedures.

3.9.2.7 Solid Radioactive Waste

3.9.2.7.a The solid radwaste system shall be used as applicable in accordance with the Process Control Program for the solidification and packaging of radioactive waste to ensure meeting the requirements of 10CFR Part 71 prior to shipment of radioactive wastes from the site.

3.9.2.7.b If the packaging requirements of 10 CFR Part 71 are not satisfied, suspend shipments of deficiently packaged solid radioactive wastes from the site until appropriate corrective measures have been taken.

<u>Basis</u>

Liquid wastes from the Radioactive Waste Disposal System are diluted in the Circulating Water System discharge prior to release to the lake.[1] With two pumps operating, the capacity of the Circulating Water System is approximately 400,000 gpm. Operation of a single circulating water pump reduces the nominal flow rate by about 50%. The circulating water flow under various operating conditions has been calculated from the head differential across the pumps and the manufacturer's head-capacity curves. Because of the low radioactivity levels in the circulating water discharge, the concentration of liquid radioactive effluents at this point is not measured directly. The concentration in the circulating water discharge is calculated from the measured concentration in the Waste Condensate Tank, the flow rate of the Waste Condensate Pumps, and the flow in the Circulating Water System. Radioactive effluents released to unrestricted areas on the basis of gross beta-gamma analysis are based on the assumption that I-129 and radium are not present. Accordingly, Appendix B, Table II, Column 2 of 10CFR20 will permit a concentration up to 1×10^{-7} uCi/ml in the circulating water discharge. Otherwise, if controlled on a radionuclide basis, the permitted discharge concentration will be in accordance with Note 1 of 10CFR20, Appendix B, Table II, Column 2. If the concentration of liquid wastes in the circulating water discharge equals the Maximum Permissible Concentration (MPC) as specified, the average concentration at the intake of the nearest public water supply at Ontario, New York, would be well below MPC.[2] Thus, these limitations provide additional assurance that the concentrations of water-borne radioactivity will result in only minimal potential public exposures within (1) Section II.A of Appendix I, 10 CFR Part 50, and (2) the limits of 10CFR Part 20.106(e).

The concentration limit for noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air was converted to an equivalent concentration in water using ICRP Publication 2 methodology.

The Specifications which limit the dose to an individual from radioactive liquid effluents are provided to implement the requirements of Sections II.A, III.A and IV.A of 10 CFR Part 50, Appendix I. The Limiting Condition for Operation implements the guides set forth in Section II.A of 10 CFR Part 50, Appendix I. The Specifications provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of 10 CFR Part 50, Appendix I. The dose calculations in the ODCM implement the requirements in Section III.A of 10 CFR Part 50, Appendix I that conformance with the guides of Appendix I is to be shown by calculational procedures based on such models and data that the actual exposure of a real individual through appropriate pathways is unlikely to be substantially underestimated. Also, there is reasonable assurance that the operation of the plant will not result in waterborne radionuclide discharges which cause the potential exposure from the finished drinking water ingestion to exceed the requirements of 40CFR 141.

The requirements that the appropriate portions of the liquid radwaste treatment system be used when specified provided assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General

Design Criterion 60 of Appendix A to 10 CFR Part 50 and design objective Section II.D of Appendix I. The limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the guide set forth in Section II.A of 10 CFR Part 50, Appendix I for liquid effluents. The cumulative maximum dose to an offsite individual from waterborne radioactive effluents is determined in order to verify that the average dose over a 31-day period is reasonably small, even if the liquid radwaste treatment system is not operated during that period. However, a cumulative dose which exceeds the stated limit does not necessarily imply that all portions of the liquid radwaste treatment system be used; certain subsystems may have only minimal effects on reducing doses.

The limit for dose rate is provided to ensure that the dose rate at any time at the site boundary from gaseous effluents will be within the annual dose limits of 10 CFR Part 20 for unrestricted areas. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of an individual in an unrestricted area, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For individuals who may at times be within the site boundary, these occupancy times will be sufficiently small to compensate for any increase in the atmospheric diffusion factor above that for the site boundary.

The Specifications which limit the dose from radioactive gaseous effluents are provided to implement the requirements of Sections II.B, II.C, III.A and IV.A of 10 CFR Part 50, Appendix I. The Limiting Condition for Operation implements the guides set forth in Sections II.B and II.C of 10 CFR Part 50, Appendix I. The Specifications provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of 10 CFR Part 50, Appendix I.

The requirement that the appropriate portions of the gaseous radwaste treatment system and the ventilation exhaust treatment system be used when specified provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and design objective Section II.D of Appendix I. The limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the guide set forth in Sections II.B and II.C of 10 CFR Part 50, Appendix I, for gaseous effluents. The cumulative maximum dose to an offsite individual from airborne radioactive effluents is determined in order to verify that the average dose over a 31-day period is reasonably small, even in the unlikely event that the gaseous radwaste treatment or ventilation exhaust systems are not operated during that period. However, a cumulative dose which exceeds the stated limit does not necessarily imply that all portions of the gaseous and ventilation exhaust treatment systems be used; certain subsystems may have only minimal effect on reducing doses.

The Specification on dose (40 CFR Part 190) is provided to meet the reporting requirements of 40 CFR Part 190. Since the plant is well removed from other fuel cycle facilities, it is sufficient to apply the Specification only to the plant in accordance with methods provided in the ODCM.

The Specification on explosive gas mixture is provided to ensure that the concentration of potentially explosive gas mixtures contained in the gas decay tanks are maintained below the flammability limit of oxygen. Maintaining the concentration of oxygen below its flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

The waste gas decay tank curie limit is provided in order to assure that in the unlikely event of an uncontrolled release of a gas decay tank's contents, the resulting total body gamma exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem.

The requirement pertaining to solid radioactive waste is provided to assure that the solid radioactive waste system will be used as appropriate for the processing and packaging of solid radioactive wastes. The specification also establishes the Process Control Program which includes the process parameters and evaluation methods used to ensure meeting the requirements of 10 CFR Part 71 prior to being shipped offsite.

References

(1) FSAR, Section 10.2

(2) FSAR, Section 2, Appendix 2A

(3) FSAR, Sections 2.6 and 2.7

3.16 **Radiological Environmental Monitoring**

Applicability

Applies to routine testing of the plant environs.

Objective

To establish a program which will assure recognition of changes in radioactivity or exposure pathways in the environs.

Specification

3.16.1 **Monitoring Program**

3.16.1.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.16-1 at the locations given in the ODCM.

3.16.1.2 If the radiological environmental monitoring program is not conducted as specified in Table 3.16-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence. (Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal availability, or to malfunction of automatic sampling equipment. If the latter, efforts shall be made to complete corrective action prior to the end of the next sampling period.)

3.16.1.3 If the level of radioactivity in an environmental sampling medium at one or more of the locations specified in the ODCM exceeds the reporting levels of Table

6.9-2 when averaged over any calendar quarter, a Special Report shall be submitted to the Commission within thirty days which includes an evaluation of any release conditions, environmental factors or other aspects which caused the reporting levels of Table 6.9-2 to be exceeded.

When more than one of the radionuclides in Table 6.9-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{limit level (1)}} + \frac{\text{concentration (2)}}{\text{limit level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 6.9-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to an individual is greater than the calendar year limit of Specifications 3.9.1.2.a or 3.9.2.2.b. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

3.16.1.4 If milk or fresh leafy vegetable samples are unavailable for more than one sample period from one or more of the sampling locations indicated by the ODCM, a discussion shall be included in the Semiannual Radioactive Effluent Report which identifies the cause of the unavailability of samples and identifies locations for obtaining replacement samples. If a milk or leafy vegetable sample location becomes unavailable, the locations from which samples were unavailable may then be deleted from the ODCM, provided that comparable locations are added to the environmental monitoring program.

3.16.2 Land Use Census

3.16.2.1 A land use census shall be conducted and shall identify the location of the nearest milk animal and the nearest residence in each of the 16 meteorological sectors within a distance of five miles.

3.16.2.2 An onsite garden located in the meteorological sector having the highest historical D/Q may be used for broad leaf vegetation sampling in lieu of a garden census; otherwise the land use census shall also identify the location of the nearest garden of greater than 500 square feet in each of the 16 meteorological sectors within a distance of five miles. D/Q shall be determined in accordance with methods described in the ODCM.

3.16.2.3 If a land use census identifies a location(s) which yields a calculated dose or dose commitment greater than that of the maximally exposed individual currently being calculated in Specification 4.12.2.2, the new identified location(s) shall be reported in the Semi-annual Radioactive Release Report.

3.16.2.4 If a land use census identifies a milk location(s) which yields a calculated dose or dose commitment greater than that at a location from which samples are currently being obtained in accordance with Specification 3.16.1, the new identified location(s) shall be reported in the Semiannual Radioactive Release Report. The new location shall be added to the radiological environmental monitoring program within thirty days, if possible. The milk location having the lowest calculated dose or dose commitment may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted.

3.16.3 Interlaboratory Comparison Program

3.16.3.1 Analyses shall be performed on applicable radioactive environmental samples supplied as part of an interlaboratory comparison program which has been approved by NRC, if such a program exists.

3.16.3.2 If analyses are not performed as required above, report the corrective actions taken to prevent a recurrence in the Annual Radiological Environmental Operating Report.

Basis

The radiological monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures of individuals resulting from the station operation. This monitoring program thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways. The initially specified monitoring program will be effective for at least three years. Following this period, program changes may be initiated based on operational experience. The detection capabilities required by Table 4.10-1 are state-of-the-art for routine environmental measurements in industrial laboratories. Lower limits of detection (LLDs) are intended as a priori (before-the-fact) limits, and analyses will be conducted in such a manner that the stated LLDs will be achieved under routine conditions.

The land use census requirement is provided to ensure that changes in the use of unrestricted areas are identified and that modifications to the monitoring program are made if required by the results of this census. A garden census is not required if an onsite garden is located in the meteorological sector having the highest historical D/Q is used for broad leaf vegetation sampling. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50.

TABLE 3.16-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Sample Locations	Sampling and Collection Frequency	Type and Frequency of Analysis
1. AIRBORNE			
a. Radioiodine	2 indicator 2 control	Continuous operation of sampler with sample collection at least once per 10 days.	Radioiodine canister. Analyze within 7 days of collection of I-131.
b. Particulates	7 indicator 5 control	Same as above.	Particulate sampler. Analyze for gross beta radioactivity $\geq$ 24 hours following filter change. Perform gamma isotopic analysis on each sample for which gross beta activity is > 10 times the mean of offsite samples. Perform gamma isotopic analysis on composite (by location) sample at least once per 92 days.
2. DIRECT RADIATION	18 indicator 10 control 11 placed greater than 5 miles from plant site	TLDs at least quarterly.	Gamma dose quarterly.

TABLE 3.16-1 (CONTINUED)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Sample Locations	Sampling and Collection Frequency	Type and Frequency of Analysis
3. WATERBORNE			
a. Surface	1 control (Russell Station) 1 indicator (Condenser Water Discharge)	Composite* sample collected over a period of ≤ 31 days.	Gross beta and gamma isotopic analysis of each composite sample. Tritium analysis of one composite sample at least once per 92 days.
b. Drinking	1 indicator (Ontario Water District Intake)	Same as above.	Same as above.

*Composite sample to be collected by collecting an aliquot at intervals not exceeding 2 hours.

TABLE 3.16-1 (CONTINUED)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Number of Samples and Sample Locations	Sampling and Collection Frequency	Type and Frequency of Analysis
4. INGESTION			
a. Milk	1 control 3 indicator June thru October each of 3 farms	At least once per 15 days.	Gamma isotopic and I-131 analysis of each sample.
	1 control 1 indicator November thru May one of the farms	At least once per 31 days.	Gamma isotopic and I-131 analysis of each sample.
b. Fish	4 control 4 indicator (Off shore at Ginna)	Twice during fishing season including at least four species.	Gamma isotopic analysis on edible portions of each sample.
c. Food Products	1 control 2 indicator (On site)	Annual at time of harvest. Sample from two of the following: 1. apples 2. cherries	Gamma isotopic analysis on edible portion of sample.
	1 control 2 indicator (On site garden or nearest offsite garden within 5 miles in the highest D/Q meteorological sector)	At time of harvest. One sample of: 1. broad leaf vegetation 2. other vegetable	Gamma isotopic analysis on edible portions of each sample.

4.0 SURVEILLANCE REQUIREMENTS

Specified intervals may be adjusted plus or minus 25% to accommodate normal test schedules.

4.1 Operational Safety Review

Applicability:

Applies to items directly related to safety limits and limiting conditions for operation.

Objective:

To specify the minimum frequency and type of surveillance to be applied to plant equipment and conditions.

Specification:

4.1.1 Calibration, testing, and checking of analog channel and testing of logic channel shall be performed as specified in Table 4.1-1.

4.1.2 Equipment and sampling tests shall be conducted as specified in Table 4.1-2 and 4.1-4.

4.1.3 Each accident monitoring instrumentation channel shall be demonstrated operable by performance of the channel check and channel calibration operations at the frequencies shown in Table 4.1-3.

4.1.4 Each radioactive effluent monitoring instrumentation channel shall be demonstrated operable by performing the channel check, source check, channel functional test, and channel calibration at the frequency shown in Table 4.1-5.

Basis:

Check

Failures such as blown instrument fuses, defective indicators, faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator action, and a check supplements this type of built-in surveillance.

Based on experience in operation of both conventional and nuclear plant systems, when the plant is in operation, the minimum checking frequency of once per shift is deemed adequate for reactor and steam system instrumentation.

Control Room procedures require a check of the Radiation Monitoring System (RMS) panel meters and strip chart recorders for proper readout once each shift. A daily surveillance log is also maintained in the Control Room for manual entry of RMS readouts, and is independently reviewed by Health Physics supervision at least weekly.

A radiation monitor downscale failure will result in a conspicuous visual indication on the RMS panel (no audible alarm). Radiation monitor control switches are spring-returned to the "operate" mode after being turned to any other test or check mode. Therefore, together with the design features of the RMS, plant surveillance procedures ensure the continued availability of each radiation monitor to perform its intended function.

Calibration

Calibrations are performed to ensure the presentation and acquisition of accurate information.

The nuclear flux (linear level) channels are calibrated daily against a heat balance standard to account for errors induced by changing rod patterns and core physics parameters.

Other channels are subject only to the "drift" errors induced within the instrumentation itself and, consequently, can tolerate longer intervals between calibration. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at intervals of each refueling shutdown. Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures. Thus, minimum calibration frequencies of once-per-day for the nuclear flux (linear level) channels, and once each refueling shutdown for the process system channels is considered acceptable.

TABLE 4.1-1 (Continued)

Channel Description	Check	Calibrate	Test	Remarks
10. Rod Position Bank Counters	S(1,2)	N.A.	N.A.	1) With analog rod position 2) Log analog rod positions each 4 hours when rod deviation monitor is out of service
11. Steam Generator Level	S	R	M	
12. Charging Flow	N.A.	R	N.A.	
13. Residual Heat Removal Pump Flow	N.A.	R	N.A.	
14. Boric Acid Tank Level	D	R	N.A.	Bubbler tube rodded weekly
15. Refueling Water Storage Tank Level	N.A.	R	N.A.	
16. Volume Control Tank Level	N.A.	R	N.A.	
17. Reactor Containment Pressure	D	R	M(1)	1) Isolation Valve signal
18. Radiation Monitoring System	D	R	M	Area Monitors R1 to R9, System Monitor R17
19. Boric Acid Control	N.A.	R	N.A.	
20. Containment Drain Sump Level	N.A.	R	N.A.	
21. Valve Temperature Interlocks	N.A.	N.A.	R	
22. Pump-Valve Interlock	R	N.A.	N.A.	
23. Turbine Trip Set-Point	N.A.	R	M(1)	1) Block Trip
24. Accumulator Level and Pressure	S	R	N.A.	

TABLE 4.1-5

Radioactive Effluent Monitoring Surveillance Requirements

Instrument	Channel Check	Source Check	Functional Test	Channel Calibration
1. Gross Activity Monitor (Liquid)				
a. Liquid Rad Waste (R-18)	D*	M(4)	Q(1)	R(5)
b. Steam Generator Blowdown (R-19)	D*	M(4)	Q(1)	R(5)
c. Turbine Building Floor Drains (R-21)	D*	M(4)	Q(1)	R(5)
d. High Conductivity Waste (R-22)	D*	M(4)	Q(1)	R(5)
e. Containment Fan Coolers (R-16)	D*	M(4)	Q(2)	R(5)
f. Spent Fuel Pool Heat Exchanger (R-20)	D*	M(4)	Q(2)	R(5)
2. Plant Ventilation				
a. Noble Gas Activity (R-14) (Alarm and Isolation of Gas Decay Tanks)	D*	M	Q(1)	R(5)
b. Particulate Sampler (R-13)	W*	N.A.	N.A.	R(5)
c. Iodine Sampler (R-10B and R-14A)	W*	N.A.	N.A.	R(5)
d. Flow Rate Determination	N.A.	N.A.	N.A.	R(6)
3. Containment Purge				
a. Noble Gas Activity (R-12)	D*	PR	Q(1)	R(5)
b. Particulate Sampler (R-11)	W*	N.A.	Q(1)	R(5)
c. Iodine Sampler (R-10A and R-12A)	W*	N.A.	N.A.	R(5)
d. Flow Rate Determination	N.A.	N.A.	N.A.	R(6)
4. Air Ejector Monitor (R-15 and R-15A)	D*	M	Q(2)	R(5)
5. Waste Gas System Oxygen Monitor	D	N.A.	N.A.	Q(3)

TABLE 4.1-5 (Continued)

TABLE NOTATION

*During releases via this pathway

(1) The Channel Functional Test shall also demonstrate that automatic isolation of this pathway and control room alarm occur if any of the following conditions exist:

1. Instrument indicates measured levels above the alarm and/or trip setpoint.

2. Power failure.

(2) The Channel Functional Test shall also demonstrate that control room alarm occurs if any of the following conditions exist:

1. Instrument indicates measured levels above the alarm setpoint.

2. Power failure.

(3) The Channel Calibration shall include the use of standard gas samples containing a nominal:

1. Zero volume percent oxygen; and

2. Three volume percent oxygen.

(4) This check may require the use of an external source due to high background in the sample chamber.

(5) Source used for the Channel Calibration shall be traceable to the National Bureau of Standards (NBS) or shall be obtained from suppliers (e.g. Amersham) that provide sources traceable to other officially-designated standards agencies.

(6) Flow rate for main plant ventilation exhaust and containment purge exhaust are calculated by the flow capacity of ventilation exhaust fans in service and shall be determined at the frequency specified.

4.10 **Radiological Environmental Monitoring**

Applicability - Applies to routine testing of plant environs.

Objective - To establish a sampling and analysis program which will assure recognition of changes in radioactivity in the environs.

Specification

4.10.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.16-1. Acceptable locations are shown in the ODCM. Samples shall be analyzed pursuant to the requirements of Tables 3.16-1 and 4.10-1.

4.10.2 A land use census shall be conducted annually (between June 1 and October 1).

4.10.3 A summary of the results obtained as part of the required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report.

Basis

The environmental survey has been designed to utilize the knowledge about dilution in the atmosphere and in the lake which has been gained during the pre-operational and operational period of study.

The radiological monitoring program provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides which lead to the highest potential radiation exposures of individuals resulting from the station operation. This monitoring program thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways.

The detection capabilities required by Table 4.10-1 are state-of-the-art for routine environmental measurements in industrial laboratories. The specified lower limits of detection for I-131 in water, milk, and other food products correspond to approximately one-quarter of the 10 CFR Part 50 Appendix I design objective dose-equivalent of 15 mrem/year for atmospheric releases and 10 mrem/year for liquid releases to the maximally exposed organ and individual.

Participation in an approved interlaboratory comparison program assures that the adequacy of environmental laboratory measurements is maintained on a continuing basis through independent cross-checking.

TABLE 4.10-1

MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD)[a]

To be achieved on 98% of analyses

Analysis	Water (pCi/l)	Airborne Particulate or Gas (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)
gross beta	4[b]	1×10^{-2}			
^{3}H	2000 (1000[b])				
^{54}Mn	15		130		
^{59}Fe	30		260		
$^{58,60}Co$	15		130		
^{65}Zn	30		260		
$^{95}Zr-Nb$	15[c]				
^{131}I	1	7×10^{-2}		1	60
$^{134,137}Cs$	15(10[b]),18	1×10^{-2}	130	15	60
$^{140}Ba-La$	15[c]			15[c]	

TABLE NOTATION

a - The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with only 5% probability of falsely concluding its presence.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda\Delta t)}$$

where

LLD is the lower limit of detection as defined above (as pCi per unit mass or volume)

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute).

E is the counting efficiency (as counts per transformation)

V is the sample size (in units of mass or volume)

2.22 is the number of transformations per minute per picocurie

Y is the fractional radiochemical yield (when applicable)

λ is the radioactive decay constant for the particular radionuclide

Δt is the elapsed time between sample collection and analysis

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background shall include the typical contributions of other radionuclides normally present in the samples (e.g., potassium-40 in milk samples). Typical values of E, V, Y and Δt should be used in the calculations.

Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidably small

TABLE 4.10-1 (Continued)

TABLE NOTATION

sample sizes, the presence of interferring nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors will be identified and described in the Annual Radiological Environmental Operating Report.

The LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as a posteriori (after the fact) limit for a particular measurement.

b - LLD for drinking water.

c - Total for parent and daughter.

4.12 **Effluent Surveillance**

Applicability

Applies to the periodic test and record requirements of the plant effluents.

Objective

To ascertain that radioactive liquid and gaseous releases from the plant are within allowable limits.

Specifications

4.12.1 Liquid Effluents

4.12.1.1 Concentration

4.12.1.1.a The radioactivity content of each batch of radioactive liquid waste to be discharged shall be determined prior to release by sampling and analysis in accordance with Table 4.12-1. The results of pre-release analyses shall be used with the calculational methods in the ODCM to assure that the concentration at the point of release is limited to the values in Specification 3.9.1.1.a.

4.12.1.1.b Post-release analyses of samples composited from batch releases shall be performed in accordance with Table 4.12-1. The results of the post-release analyses shall be used with the calculational methods in the ODCM to assure that the does commitments from liquids were limited to the values in Specification 3.9.1.2.a.

4.12.1.2 Dose; Liquid Waste Treatment

4.12.1.2.a Cumulative dose contributions from liquid effluents shall be determined in accordance with the ODCM at least once per 31 days.

4.12.2 Gaseous Wastes

4.12.2.1 Release Rate

4.12.2.1.a The gas effluent continuous monitors as listed in Table 3.5-6 having provisions for the automatic termination of gas decay tank or containment purge releases, shall be used to limit releases within the values established in Specification 3.9.2.1 when monitor setpoint values are exceeded.

4.12.2.1.b The dose rate due to radioactive materials, other than noble gases, in gaseous effluents shall be determined in accordance with the methods of the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program, specified in Table 4.12-2.

4.12.2.2 Dose (10 CFR Part 50, Appendix I); Gaseous Waste Treatment

4.12.2.2.a Cumulative dose contributions from gaseous effluents shall be determined in accordance with the ODCM at least once every 31 days.

4.12.3 Waste Gas Decay Tanks

The quantity of radioactive material contained in each waste gas decay tank shall be determined to be within the limit specified in 3.9.2.6.a at least once per 24 hours if the total primary coolant noble gas concentration exceeds 250 μCi/gram and primary coolant gas is being transferred to the gaseous radwaste treatment system.

Basis:

Sufficient tests will be made to be certain that radioactive materials are not released to the environment in quantities greater than allowable. Installed radiation monitoring equipment in the plant will be used in conjunction with laboratory analyses to maintain surveillance of normal effluents.

Sufficient records will be maintained to determine the concentration of radioactive materials in unrestricted areas. Isotopic analysis of representative samples will serve to verify the accuracy of routine samples by identification of significant energy peaks.

The quantity of radioactivity in each gas decay tank is determined when the noble gas concentration in the primary coolant system increases significantly enough to potentially contribute an appreciable quantity of noble gas activity to the gaseous radwaste system. The required surveillance will be initiated at a primary noble gas concentration level which, if attained will still allow sufficient margin below the specified curie limit for a single gas decay tank. Determination of tank curie content may be performed by sampling and/or calculation.

TABLE 4.12-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (uCi/ml)[a]
Batch Waste Release Tanks[b]	PR Each Batch	PR Each Batch	1. Principal Gamma Emitters[d] and I-131 or 2. Gross beta-gamma*	5×10^{-7} 1×10^{-6} 5×10^{-7}
	PR One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	PR Each Batch	M Composite[c]	H-3 Gross alpha	1×10^{-5} 1×10^{-7}
	PR Each Batch	Q Composite[c]	Sr-89, Sr-90 Fe-55	5×10^{-8} 1×10^{-6}
Continuous Release[e]				
Retention Tank	Continuous	W Composite[c]	Principal Gamma Emitters[d] and I-131	5×10^{-7} 1×10^{-6}
Service Water (CV Fan Cooler and SFP HX lines)	Continuous	M or S** Grab	Gross beta-gamma	1×10^{-7}

* If gross beta is performed for batch releases, then a weekly composite shall also be analyzed for Principal Gamma Emitters and I-131.

**Service water samples shall be taken and analyzed once per 12 hours if alarm setpoint is reached on continuous monitor.

TABLE NOTATION

a. The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with 5% probability of falsely concluding its presence.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

where

LLD is the lower limit of detection as defined above (as uCi per unit mass or volume)

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute).

E is the counting efficiency (as counts per transformation)

V is the sample size (in units of mass or volume)

2.22×10^6 is the number of transformations per minute per microcurie

Y is the fractional radiochemical yield (when applicable)

λ is the radioactive decay constant for the particular radionuclide

Δt is the elapsed time between midpoint of sample collection and time of counting (for plant effluents, not environmental samples).

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background shall include the typical contributions of other radionuclides normally present in the samples. Typical values of E, V, Y and Δt should be used in the calculation.

The background count rate is calculated from the background counts that are determined to be within ± one FWHM energy band about the energy of the gamma ray peak used for the quantitative analysis for this radionuclide.

The LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as a posteriori (after the fact) limit for a particular measurement.

Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidably small sample sizes, the presence of interferring nuclides, or other uncontrollable circumstances may render these LLDs unachievable. When circumstances result in LLDs higher than required, the reasons shall be documented in the Semiannual Radioactive Effluent Report.

b. A batch release is the discharge of liquid wastes of a discrete volume.

c. A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen which is representative of the liquids released.

d. The principal gamma emitters for which the LLD specification will apply are exclusively the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Cs-134, Cs-137, and Ce-141. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported. Nuclides which are below the LLD for the analyses should be reported as less than the LLD and should not be reported as being present at the LLD level. The less than values should not be used in the required dose calculations. When unusual circumstances result in LLDs higher than required, the reasons shall be documented in the Semiannual Radioactive Effluent Release Report.

e. A continuous release is the discharge of liquid wastes of a non-discrete volume; e.g. from a volume of system that has an input flow during the continuous release.

TABLE 4.12-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (uCi/ml)[a]
A. Gas Decay Tank	PR Each Tank Grab Sample	PR Each Tank	Principal Gamma Emitters[e]	1×10^{-4}
B. Containment Purge	PR Each Purge[b,c] Grab Sample	PR Each Purge[b]	Principal Gamma Emitters[e]	1×10^{-4}
			H-3	1×10^{-6}
C. Auxiliary Building Ventilation	M[b] Grab Sample	M[b]	Principal Gamma Emitters[e]	1×10^{-4}
			H-3	1×10^{-6}
D. Air Ejector	M[b,f,h] Grab Sample	M[b]	Principal Gamma Emitters[e], I-131	1×10^{-4}
			H-3[g]	1×10^{-6}
E. All Release Types as listed in B and C above	Continuous[d]	W[b] Charcoal Sample	I-131	1×10^{-12}
			I-133	1×10^{-10}
	Continuous[d]	W[b] Particulate Sample	Principal Gamma Emitters[e] (I-131, Others)	1×10^{-11}
	Continuous[d]	M Composite Particulate Sample	Gross alpha	1×10^{-11}
	Continuous[d]	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
F. All Release Types as listed in B,C and D above	Continuous[d]	Noble Gas Monitor	Beta or Gamma	1×10^{-6}

TABLE 4.12-2 (Continued)

TABLE NOTATION

a. The lower limit of detection (LLD) is defined in Table Notation a. of Table 4.12-1.

b. Analyses shall also be performed when the monitor on the continuous sampler reaches its setpoint.

c. Tritium grab samples shall be taken at least three times per week when the reactor cavity is flooded.

d. The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with specifications 3.9.2.1.a, 3.9.2.2.a and 3.9.2.2.b.

e. The principal gamma emitters for which the LLD specification will apply are exclusively the following radionuclides: Kr-85m, Xe-133, Xe-133m, and Xe-135 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported. Nuclides which are below the LLD for the analyses should not be reported as being present at the LLD level for that nuclide. When unusual circumstances result in LLDs higher than required, the reasons shall be documented in the Semiannual Effluent Release Report.

f. Air ejector samples are not required during cold or refueling shutdowns.

g. Air ejector tritium sample not required if the secondary activity is less than 1×10^{-4} μCi/gm.

h. Air ejector iodine samples shall be taken and analyzed weekly if the secondary coolant activity exceeds 1×10^{-4} μCi/gm.

5.0 DESIGN FEATURES

5.1 Site

The R. E. Ginna Nuclear Power Plant is located on property owned by Rochester Gas and Electric Corporation at a site on the south shore of Lake Ontario, approximately 16 miles east of Rochester, New York. The site map shown in Figure 5.1-1 depicts the Ginna Exclusion Area Boundary and Site Boundary locations. For purposes of implementing Ginna Radiological Technical Specifications, and for evaluating radiological releases to the Unrestricted Area, the Unrestricted Area Boundary is assumed to coincide with the Exclusion Area Boundary.

GINNA SITE MAP

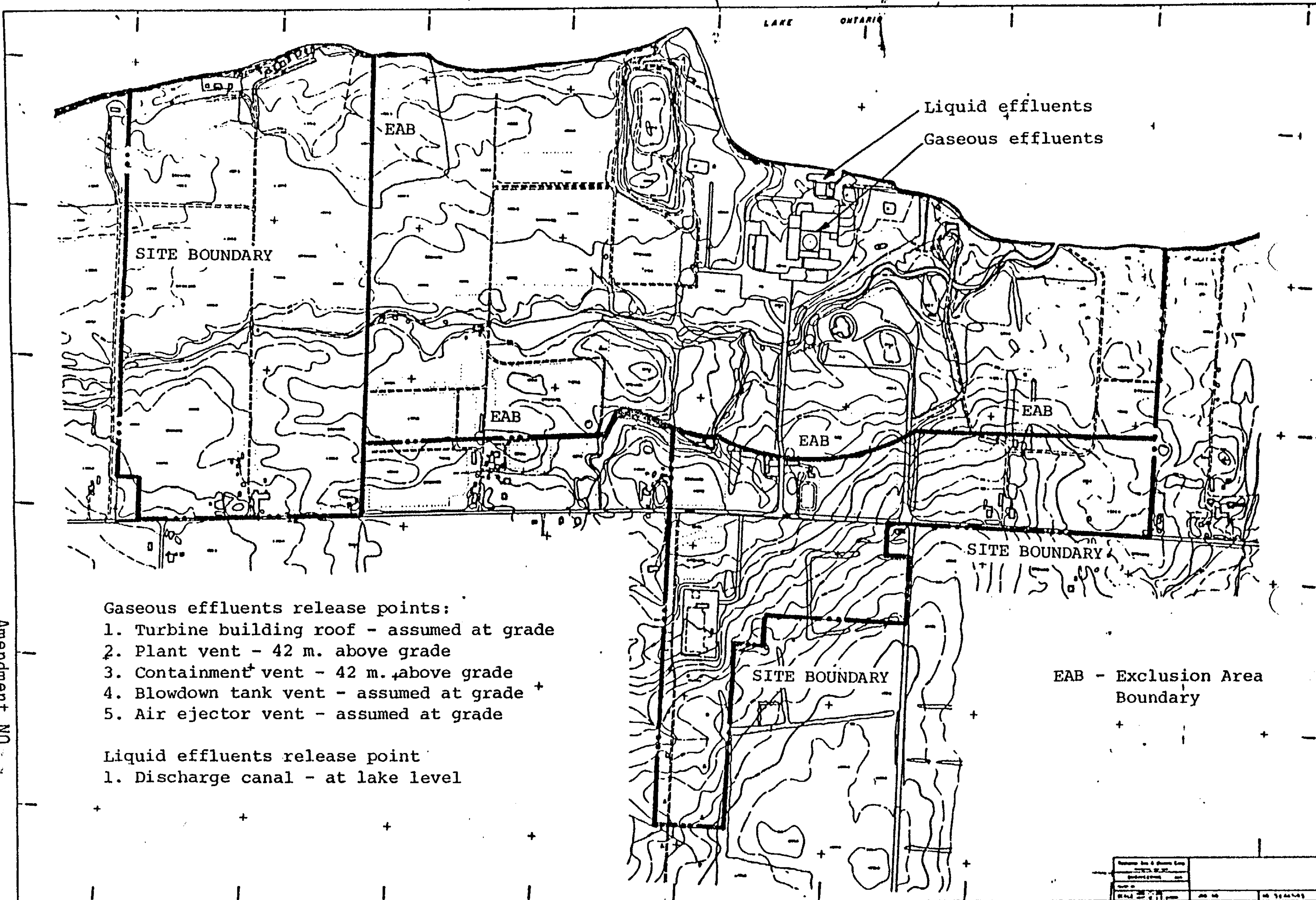

FIGURE 5.1-1

5.5 Waste Treatment Systems

5.5.1 Radioactive Liquid Waste Treatment

The liquid waste treatment system consists of a Waste Holdup Tank, a Waste Evaporator and a mixed bed demineralizer. Portions of the system may be bypassed and still meet the release limits.

5.5.2 Gaseous Radwaste Treatment

The gaseous radwaste system is designed to collect off-gas from the primary coolant system and hold for radioactive decay prior to release to the environment.

The gaseous radwaste treatment system consists of four (4) Gas Decay Tanks and two (2) gas compressors. Only one compressor and three Gas Decay Tanks are necessary to the system.

5.5.3 Ventilation Exhaust System

The ventilation exhaust is treated to reduce gaseous radioiodine and material in particulate form by passing through charcoal adsorbers and/or HEPA filters. This system has no effect on noble gas effluents. The components of the ventilation exhaust system are:

- Auxiliary Building HEPA filters
- Auxiliary Building "G" Charcoal & HEPA filters
- Auxiliary Building "A" Charcoal Adsorbers
- Containment Purge Charcoal & HEPA filters

5.5.4 Solid Radwaste System

The solid radwaste system consists of piping and valves in the Drumming Station whereby waste evaporator concentrates are transferred into prepared drums by means of the waste evaporator feed pump. Alternatively, liquid wastes may be solidified and prepared for shipment by a contractor.

AUDITS (Continued)

g. The Facility Fire Protection Program and implementing procedures at least once per two years.

h. An independent fire protection and loss prevention program inspection and audit performed at least once per 12 months utilizing either qualified offsite licensee personnel or an outside fire protection firm.

i. An inspection and audit of the fire protection and loss prevention program performed by non-licensee personnel at least once per 36 months. The personnel may be representatives of ANI, an insurance brokerage firm, or other qualified individuals.

j. The radiological environmental monitoring program and the results thereof at least once per 12 months.

k. The Offsite Dose Calculation Manual and implementing procedures at least once per 24 months.

l. The Process Control Program and implementing procedures at least once per 24 months.

m. Any other area of facility operation considered appropriate by the NSARB or the Vice President, Electric and Steam Production.

AUTHORITY

6.5.2.9 a. The chairman of the Nuclear Safety Audit and Review Board is responsible to the Executive Vice President on all activities for which the review board is responsible.

b. The NSARB shall report to and advise the Vice President, Electric and Steam Production, on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.2.8.

RECORDS

6.5.2.10 Records of NSARB activities shall be prepared, approved, and distributed as indicated below:

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November 1972.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. The radiological environmental monitoring program.

h. Offsite Dose Calculation Manual implementation.

i. Process Control Program implementation.

6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, shall be reviewed by the PORC and approved by the Station Superintendent prior to implementation and reviewed periodically as set forth in the applicable procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedures is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom is the Shift Foreman who holds a Senior Reactor Operator's License.

c. The change is documented, reviewed by the PORC, and approved by the Station Superintendent within 10 days of implementation.

6.9 **Reporting Requirements**

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Regional Administrator of the USNRC, Region 1, unless otherwise noted.

6.9.1 Routine Reports

6.9.1.1 Startup Report. A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests performed and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

Startup reports shall be submitted within (1) 90 days following completion of the startup test program, or (2) 90 days following resumption of commercial power operation, whichever is earliest. If the Startup Report does not cover both events (i.e., completion of startup test program, and resumption of commercial power operation), supplementary reports shall be submitted at least every three months until both events have been completed.

6.9.1.2 Monthly Operating Report. Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office of Management Information and Program Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 by the fifteenth of each month following the calendar month covered by the report. The monthly report shall include a narrative summary of operating experience describing the operation of the facility, including major safety related maintenance for the monthly period, except that safety related maintenance performed during the refueling outage may be reported in the monthly report for the month following the end of the outage rather than each month during the outage.

6.9.1.3 Annual Radiological Environmental Operating Report

A radiological environmental operating report covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year.

The annual radiological environmental report shall include summaries, interpretations, and analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with background (control) samples and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses as required.

The annual radiological environmental operating report shall include summarized and tabulated results in the format of Table 6.9-1 of all radiological environmental samples taken during the report period. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report. In addition, the annual report shall include a discussion which identifies the circumstances which prevent any required detection limits for environmental sample analyses from being met, and a discussion of all deviations from the sample schedule of Table 3.16-1. The report shall also include the following: a summary description of the radiological environmental monitoring program including a map of all sampling locations keyed to a table giving distances and directions from the reactor, and the results of the participation in an interlaboratory comparison program.

6.9.1.4 Semiannual Radioactive Effluent Release Report

Routine radioactive effluent release reports covering the operation of the unit during the previous six months of operation shall be submitted within 60 days after January 1 and July 1 of each year. This report shall include a summary, on a quarterly basis, of the quantities of radioactive liquid and gaseous effluents and solid waste released as outlined in Regulatory Guide 1.21, Revision 1.

The radioactive effluent release report submitted within 60 days of January 1 shall include an assessment of radiation doses from the radioactive liquid and gaseous effluents released from the unit during each of the previous four calendar quarters as outlined in Regulatory Guide 1.21, Revision 1. In addition, the site boundary maximum noble gas gamma air and beta air doses shall be evaluated. The assessment of radiation doses shall be performed in accordance with the ODCM. This same report shall include an annual summary of hourly meteorological data collected over the previous calendar year. Alternatively, the licensee has the option of retaining this summary on site in a file that shall be provided to the NRC upon request.

Also, the semiannual report shall include any new location(s) identified by the land use census which yield a calculated does or dose commitment greater than those forming the basis of Specifications 4.12.2.2 or 3.16.1. The report shall also contain a discussion which identifies the causes of the unavailability of milk or leafy vegetable samples and identifies locations for obtaining replacement samples in accordance with specification 3.16.1.4.

The radioactive effluent release report shall include a discussion which identifies the circumstances which prevent any required detection limits for effluent sample analyses from being met.

The radioactive effluent release reports shall include any changes made during the reporting period to the ODCM as specified in Section 6.15, and to the Process Control Program as specified in Section 6.16. The radioactive effluent release reports shall also include a discussion of any major changes to radioactive waste treatment systems in accordance with Specification 6.17.2.1.

6.9.2 Reportable Occurrences

Reportable occurrences, including corrective actions and measures to prevent reoccurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

6.9.3 Unique Reporting Requirements

6.9.3.1 Annually: Results of required leak tests performed on sources if the tests reveal the presence of 0.005 microcurie or more of removable contamination.

6.9.3.2 Annually: A tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions, e.g., reactor operations and surveillance, in-service inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions. (NOTE: This tabulation supplements the requirements of Section 20.407 of 10 CFR Part 20.)

6.9.3.3 Reactor Overpressure Protection System Operation

In the event either the PORVs or the RCS vent(s) are used to mitigate a RCS pressure transient, a Special Report shall be prepared and submitted to the Commission within thirty days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or vent(s) on the transient and any other corrective action necessary to prevent recurrence.

6.9.3.4 Special reports shall be submitted to the Director of the NRC Regional Office listed in Appendix D, 10 CFR Part 20, with a copy to the Director, Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555 within the time period specified for each report.

TABLE 6.9-1

ENVIRONMENTAL RADIOLOGICAL MONITORING PROGRAM SUMMARY

Name of Facility R. E. Ginna Nuclear Power Plant Docket No. 50-244

Location of Facility Wayne County, New York Reporting Period ____________________

Medium or Pathway Sampled (Unit of Measurement	Type and Total Number of Analyses Performed	Lower Limit of Detection[a] (LLD)	All Indicator Locations Mean (1)[b] Range[b]	Locations with Highest Annual Mean Name Distance and Direction	Mean(1)[b] Range[b]	Control Locations Mean (1)[b] Range[b]

[a]Nominal Lower Limit of Detection (LLD) as defined in Table Notation a. of Table 4.12-1.

[b]Mean and range based upon detectable measurements only. Fraction of detectable measurements at specified locations is indicated in parentheses (1).

TABLE 6.9-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/Kg, wet)	Milk (pCi/l)	Broad Leaf Vegetables (pCi/Kg, wet)
H-3	2×10^4				
Mn-54	1000		3×10^4		
Fe-59	400		1×10^4		
Co-58	1000		3×10^4		
Co-60	300		1×10^4		
Zn-65	300		2×10^4		
Zr-Nb-95	400[a]				
I-131	2	0.9		3	1×10^2
Cs-134	30	10	1×10^3	60	1×10^3
Cs-137	50	20	2×10^3	70	2×10^3
Ba-La-140	200[a]			300	

(a) Total for parent and daughter

6.15 <u>Offsite Dose Calculation Manual</u> (ODCM)

6.15.1 Any changes to the ODCM shall be made by the following method:

6.15.1.a Licensee initiated changes shall be submitted to the Commission with the Semi-annual Radioactive Effluent Release Report for the period in which the change(s) was made and shall contain:

(i) sufficiently detailed information to support the rationale for the change.

(ii) a determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

(iii) documentation of the fact that the change has been reviewed and found acceptable by the PORC.

6.15.1.b Licensee initiated changes shall become effective after review and acceptance by the PORC on a date specified by the licensee.

6.16 **Process Control Program** (PCP)

6.16.1 Any changes to the PCP shall be made by the following method:

6.16.1.a Licensee initiated changes shall be submitted to the Commission with the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made and shall contain:

(i) sufficiently detailed information to support the rationale for the change;

(ii) a determination that the change will not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

(iii) documentation of the fact that the change has been reviewed and found acceptable by the PORC.

6.16.1.b Licensee initiated changes shall become effective after review and acceptance by the PORC on a date specified by the licensee.

6.17 **Major Changes to Radioactive Waste Treatment Systems (Liquid, Gaseous and Solid)**

FUNCTION

6.17.1 The radioactive waste treatment systems (liquid, gaseous and solid) are those systems defined in Technical Specification 5.5.

6.17.2 Major changes to the radioactive waste systems (liquid and gaseous) shall be reported by the following method. For the purpose of this specification, "major changes" is defined in Specification 6.17.3 below.

6.17.2.1 The Commission shall be informed of all major changes by the inclusion of a suitable discussion or by reference to a suitable discussion of each change in the Semiannual Radioactive Effluent Release Report for the period in which the changes were made. The discussion of each change shall contain:

a) a summary of the evaluation that led to the determination that the change could be made (in accordance with 10 CFR 50.59);

b) sufficient detailed information to support the reason for the change;

c) a detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

d) an evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents from those previously predicted;

e) an evaluation of the change which shows the expected maximum exposures to individual in the unrestricted area and to the general population from those previously estimated;

f) documentation of the fact that the change was reviewed and found acceptable by the PORC.

6.17.3 "Major Changes" to radioactive waste systems (liquid, gaseous and solid) shall include the following:

a) Major changes in process equipment, components, and structures from those in use (e.g., deletion of evaporators and installation of demineralizers);

b) Major changes in the design of radwaste treatment systems (liquid, gaseous and solid) that could significantly alter the characteristics and/or quantities of effluents released;

c) Changes in system design which may invalidate the accident analysis (e.g., changes in tank capacity that would alter the curies released).

SAFETY EVALUATION
BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT NO. TO PROVISIONAL OPERATING LICENSE NO. DPR-18
R. E. GINNA NUCLEAR POWER PLANT
DOCKET NO. 50-244

1.0 INTRODUCTION

To comply with Section V of Appendix I of 10 CFR Part 50, the Rochester Gas and Electric Corporation has filed with the Commission plans and proposed technical specifications developed for the purpose of keeping releases of radioactive materials to unrestricted areas during normal operations, including expected operational occurrences, as low as is reasonably achievable. The Rochester Gas and Electric Corporation filed this information with the Commission by letter dated August 12, 1982* which requested changes to the Technical Specifications appended to Provisional Operating License No. DPR-18 for R. E. Ginna Nuclear Power Plant. The proposed technical specifications update those portions of the technical specifications addressing radioactive waste management and make them consistent with the current staff positions as expressed in NUREG-0472. These revised technical specifications would reasonably assure compliance, in radioactive waste management, with the provisions of 10 CFR Part 50.36a, as supplemented by Appendix I to 10 CFR Part 50, with 10 CFR Parts 20.105(c), 106(g), and 405(c); with 10 CFR Part 50, Appendix A, General Design Criteria 60, 63, and 64; and with 10 CFR Part 50, Appendix B.

*Submittals by Rochester Gas and Electric Corporation dated 02/14/79, 05/29/79, 01/10/83 and 03/04/83 also relate to this evaluation.

2.0 BACKGROUND AND DISCUSSION

2.1 Regulations

10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities", Section 50,36a, "Technical Specifications on Effluents from Nuclear Power Reactors", provides that each license authorizing operation of a nuclear power reactor will include technical specifications that (1) require compliance with applicable provisions of Part 20.106, "Radioactivity in Effluents to Restricted Areas"; (2) require that operating procedures developed for the control of effluents be established and followed; (3) require that equipment installed in the radioactive waste system be maintained and used; and (4) require the periodic submission of reports to the NRC specifying the quantity of each of the principal radionuclides released to unrestricted areas in liquid and gaseous effluents, any quantities of radioactive materials released that are significantly above design objectives, and such other information as may be required by the Commission to estimate maximum potential radiation dose to the public resulting from the effluent releases.

10 CFR Part 20, "Standards for Protection Against Radiation," paragraphs 20.105(c), 20.106(g), and 20.405(c), require that nuclear power plant and other licensees comply with 40 CFR Part 190," "Environmental Radiation Protection Standards for Nuclear Power Operations" and submit reports to the NRC when the 40 CFR Part 190 limits have been or may be exceeded.

2.1 10 CFR Part 50, Appendix A - General Design Criteria for Nuclear Power Plants, contains Criterion 60, Control of releases of radioactive materials to the environment; Criterion 63, Monitoring fuel and waste storage; and Criterion 64, Monitoring radioactivity releases. Criterion 60 requires that the nuclear power unit design include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Criterion 63 requires that appropriate systems be provided in radioactive waste systems and associated handling areas to detect conditions that may result in excessive radiation levels and to initiate appropriate safety actions. Criterion 64 requires that means be provided for monitoring effluent discharge paths and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences and postulated accidents.

10 CFR Part 50, Appendix B, establishes quality assurance requirements for nuclear power plants.

10 CFR Part 50, Appendix I, Section IV, provides guides on technical specifications for limiting conditions for operation for light-water-cooled nuclear power reactors licensed under 10 CFR Part 50.

2.2 Standard Radiological Effluent Technical Specifications

NUREG-0472 provides radiological effluent technical specifications for pressurized water reactors which the staff finds to be an acceptable standard for licensing actions. Further clarification of these acceptable methods is provided in NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants." NUREG-0133 describes methods found acceptable to the staff of the NRC for the calculation of certain key values required in the preparation of proposed radiological effluent technical specifications for light-water-cooled nuclear power plants. NUREG-0133 also provides guidance to licensees in preparing requests for changes to existing radiological effluent technical specifications for operating reactors. It also describes current staff positions on the methodology for estimating radiation exposure due to the release of radioactive materials in effluents and on the administrative control of radioactive waste treatment systems.

The above NUREG documents address all of the radiological effluent technical specifications needed to assure compliance with the guidance and requirements provided by the regulations previously cited. However, alternative approaches to the preparation of radiological effluent technical specifications and alternative radiological effluent technical specifications may be acceptable if the staff determines that the alternatives are in compliance with the regulations and with the intent of the regulatory guidance.

2.2 The standard radiological effluent technical specifications can be grouped under the following categories:

(1) Instrumentation
(2) Radioactive effluents
(3) Radiological environmental monitoring
(4) Design features
(5) Administrative controls

Each of the specifications under the first three categories are comprised of two parts: the limiting condition for operation and the surveillance requirements. The limiting condition for operation provides a statement of the limiting condition, the times when it is applicable, and the actions to be taken in the event that the limiting condition is not met.

In general, the specifications established to assure compliance with 10 CFR Part 20 standards provide, in the event the limiting conditions of operation are exceeded, that without delay conditions are restored to within the limiting conditions. Otherwise, the facility is required to effect approved shutdown procedures. In general, the specifications established to assure compliance with 10 CFR Part 50 provide, in the event the limiting conditions of operation are exceeded, that within specified times corrective actions are to be taken, alternative means of operation are to be employed, and certain reports are to be submitted to the NRC describing these conditions and actions.

The specifications concerning design features and administrative controls contain no limiting conditions of operation or surveillance requirements.

2.2 Table 1 indicates the standard radiological effluent technical specifications that are needed to assure compliance with the particular provisions of the regulations described in Section 1.0.

Table 1. Relation Between Provisions of the Regulations and the Standard Radiological Effluent Technical Specifications for Pressurized Water Reactors and Boiling Water Reactors

● Indicate the specifications that are needed to assure compliance with the identified provision of the regulations.

	Standard Radiological Effluent Technical Specifications																												
	Instrumentation		Radioactive Effluents																Rad. Envir. Monitoring			Design Features	Administrative Control						
			Liquid				Gaseous																						
							PWR/BWR				PWR		BWR																
Provisions of Title 10 Code of Federal Regulations	Rad. Liquid Effl. Monitoring	Rad. Gas. Effl. Monitoring	Effluent Concentration	Dose	Liquid Radwaste Treatment	Liquid Holdup Tanks	Dose Rate	Dose Noble Gases	Dose I-131, Trit. and Part.	Explosive Gas Mixture	Gaseous Radwaste Treatment	Gas Storage Tanks	Gaseous Radwaste Treatment	Ventilation Exhaust Treatment	Main Condenser	Mark I or II Containment	Solid Radioactive Waste	Total Dose	Rad. Env. Monitoring Program	Land Use Census	Interlab. Comparison Program	Site Boundaries*	Review and Audits	Procedures	Reports	Record Retention	Process Control Program	Offsite Dose Calc. Manual	Major Changes to Rad. Systems
§ 50.36a Technical specifications on effluents from nuclear power reactors																													
Remain within limits of § 20.106			●			●	●					●				●												●	
Establish and follow procedures to control effluents			●	●	●	●	●	●	●	●	●	●	●	●	●	●	●							●				●	
Maintain and use radioactive waste system equipment				●	●			●	●		●		●	●														●	
Submit reports, semi-annual and other	●	●	●	●	●	●	●	●	●	●	●	●	●	●			●	●									●	●	●
§§ 20.105(c), 20.106(g), 20.405(c) Compliance with 40 CFR 190																		●	●	●	●							●	
Part 50 Appendix A - General Design Criteria																													
Criterion 60 - Control of releases of radioactive materials to the environment	●	●			●	●				●	●	●	●	●	●	●	●							●			●	●	
Criterion 61 - Fuel storage and handling and radioactivity control	●	●																											
Criterion 63 - Monitoring fuel and waste storage	●	●																											
Criterion 64 - Monitoring radioactivity releases	●	●													●				●									●	
Part 50 Appendix B - Quality Assurance Criteria	●	●																			●		●	●			●		
Part 50 Appendix I - Guides to Meet "As Low As Is Reasonably Achievable (ALARA)"																													
Maintain releases within design objectives				●	●			●	●		●			●														●	
Establish surveillance & monitoring program to provide data on:																													
(1) quantities of rad. matls. in effluents	●	●																											
(2) radiation & rad. matls. in the environment																			●						●	●			
(3) changes in use of unrestricted areas																				●									
Exert best efforts to keep releases "ALARA"				●	●			●	●		●		●	●															●
Submit report if calculated doses exceed the design objective				●				●	●		●			●											●				
Demonstrate conform. to des. obj. by calc. proced.				●				●	●		●			●														●	
Part 100															●														

*Note: Needed to fully implement other specifications.

-5-

3.0 EVALUATION

The enclosed report (TER-C5506-117) was prepared by Franklin Research Center (FRC) as part of our technical assistance contract program. Their report provides their technical evaluation of the compliance of the licensee's submittal with NRC provided criteria. We have reviewed the FRC report and concur with the conclusions therein.

3.1 SUMMARY

The proposed changes to the radiological effluent technical specifications for the R. E. Ginna Nuclear Power Plant have been evaluated, reviewed, and found to be in compliance with the requirements of the NRC regulations and with the intent of NUREG-0133 and NUREG-0472 (the Ginna plant is comprised of one pressurized water reactor) and thereby fulfill all the requirements of the regulations related to radiological effluent technical specifications.

The proposed changes would not remove or relax any existing requirement related to the probability or consequences of accidents previously considered and do not involve a significant hazards consideration.

The proposed changes would not remove or relax any existing requirement needed to provide reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner. The staff therefore, finds the proposed changes acceptable.

4.0 ENVIRONMENTAL CONSIDERATIONS

We have determined that the issuance of the proposed amendment to the Technical Specifications appended to Provisional Operating License No. DPR-18 for R. E. Ginna Nuclear Power Plant would not authorize a significant change in the types, or a significant increase in the amounts, of effluents or in the authorized power level, and that the amendment will not result in any significant environmental impact. Having made these determinations, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR Part 51.5(d)(4), that environmental impact statement or negative declaration, and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

5.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or the health and safety of the public.

6.0 ACKNOWLEDGEMENT

W. Meinke contributed to this evaluation.

Attachment: TER dated 01/19/83

Date: September 28, 1983

The requirement for participation in an interlaboratory comparison program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of a quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid. Only samples with radioactivity levels comparable to levels in environmental samples need be analyzed.

TECHNICAL EVALUATION REPORT

RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATION IMPLEMENTATION (A-2)

ROCHESTER GAS AND ELECTRIC CORPORATION
R. E. GINNA NUCLEAR POWER PLANT

NRC DOCKET NO. 50-244

NRC TAC NO. 8097

NRC CONTRACT NO. NRC-03-81-130

FRC PROJECT C5506

FRC ASSIGNMENT 4

FRC TASK 93

Prepared by

Franklin Research Center
20th and Race Streets
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Author: S. Chen

FRC Group Leader: S. Pandey

Prepared for

Nuclear Regulatory Commission
Washington, D.C. 20555

Lead NRC Engineer: F. Congel
C. Willis

January 19, 1983

Franklin Research Center
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The Benjamin Franklin Parkway, Phila., Pa. 19103 (215) 448-1000

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January 19, 1983

Prepared by:	Reviewed by:	Approved by:
S. Y. Chen	S. Pandey	[illegible] for S. Carfagno
Principal Author	Group Leader	Department Director/
Date: 1/19/83	Date: 1/19/83	Date: 1/19/83

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FOREWORD

This Technical Evaluation Report was prepared by Franklin Research Center under a contract with the U.S. Nuclear Regulatory Commission (Office of Nuclear Reactor Regulation, Division of Operating Reactors) for technical assistance in support of NRC operating reactor licensing actions. The technical evaluation was conducted in accordance with criteria established by the NRC.

1. INTRODUCTION

1.1 PURPOSE OF REVIEW

The purpose of this technical evaluation report (TER) is to review and evaluate the proposed changes in the Technical Specifications of R. E. Ginna Nuclear Power Plant with regard to Radiological Effluent Technical Specifications (RETS) and the Offsite Dose Calculation Manual (ODCM).

The evaluation uses criteria proposed by the NRC staff in the Model Technical Specifications for pressurized water reactors (PWRs), NUREG-0472 [1]. This effort is directed toward the NRC objective of implementing RETS which comply principally with the regulatory requirements of the Code of Federal Regulations, Title 10, Part 50 (10CFR50), "Domestic Licensing of Production and Utilization Facilities," Appendix I [2]. Other regulations pertinent to the control of effluent releases are also included within the scope of compliance.

1.2 GENERIC BACKGROUND

Since 1970, 10CFR50, Section 50.36a, "Technical Specifications on Effluents from Nuclear Power Reactors," has required licensees to provide technical specifications which ensure that radioactive releases will be kept as low as reasonably achievable (ALARA). In 1975, numerical guidance for the ALARA requirement was issued in 10CFR50, Appendix I [3]. The licensees of all operating reactors were required to submit, no later than June 4, 1976, their proposed ALARA Technical Specifications and information for evaluation in accordance with 10CFR50, Appendix I.

However, in February 1976, the NRC staff recommended that proposals to modify Technical Specifications be deferred until the NRC completed the model RETS. The model RETS deals with radioactive waste management systems and environmental monitoring. Although the model RETS closely parallels 10CFR50, Appendix I requirements, it also includes provisions for addressing other issues.

These other issues are specifically stipulated by the following regulations:

- o 10CFR20 [4], "Standards for Protection Against Radiation," Paragraphs 20.105(c), 20.106(g), and 20.405(c) require that nuclear power plants and other licensees comply with 40CFR190 [5], "Environmental Radiation Protection Standards for Nuclear Power Operations," and submit reports to the NRC when the 40CFR190 limits have been or may be exceeded.
- o 10CFR50, Appendix A [6], "General Design Criteria for Nuclear Power Plants," contains Criterion 60 - Control of releases of radioactive materials to the environment; Criterion 63 - Monitoring fuel and waste storage; and Criterion 64 - Monitoring radioactivity releases.
- o 10CFR50, Appendix B [7], establishes the quality assurance required for nuclear power plants.

The current NRC position on the model RETS was established in May 1978 when the NRC's Regulatory Requirements Review Committee approved the model RETS: NUREG-0472 for PWRs [1] and NUREG-0473 [8] for boiling water reactors (BWRs). Copies were sent to licensees in July 1978 with a request to submit proposed site-specific RETS on a staggered schedule over a 6-month period. Licensees responded with requests for clarifications and extensions.

The Atomic Industrial Forum (AIF) formed a task force to comment on the model RETS. NRC staff members first met with the AIF task force on June 17, 1978. The model RETS was subsequently revised to reflect comments from the AIF and others. A principal change was the transfer of much of the material concerning dose calculations from the model RETS to a separate ODCM.

The revised model RETS was sent to licensees on November 15 and 16, 1978 with guidance (NUREG-0133 [9]) for preparation of the RETS and the ODCM and a new schedule for responses, again staggered over a 6-month period.

Four regional seminars on the RETS were conducted by the NRC staff during November and December 1978. Subsequently, Revision 2 of the model RETS and additional guidance on the ODCM and a Process Control Program (PCP) were issued in February 1979 to each utility at individual meetings. In response to the NRC's request, operating reactor licensees subsequently submitted initial proposals on plant RETS and the ODCM. Review leading to ultimate implementation of these documents was initiated by the NRC in 1981 using subcontracted independent teams as reviewers.

As the RETS review process has progressed since September 1981, feedback from the licensees has led the NRC to believe that modification to some of the guidelines in the current version of Revision 2 is needed to clarify specific concerns of the licensees and thus expedite the entire review process. Starting in April 1982, NRC distributed revised versions of RETS in draft form to the licensees during site visits. The new guidance on these changes was presented in the AIF meeting on May 19, 1982 [10]. Some interim changes regarding the Radiological Environmental Monitoring Section were issued in August 1982 [11]. With the incorporation of these new changes, NRC issued, in September 1982, a draft version of NUREG-0472, Revision 3 [12], to serve as new guidance for the review teams.

1.3 PLANT-SPECIFIC BACKGROUND

In response to the NRC's request, the Licensee, Rochester Gas and Electric Corporation (RG&E), submitted a RETS proposal dated February 14, 1979 [13] on behalf of R. E. Ginna Nuclear Power Plant. This proposal also included the ODCM [14]. In the RETS submittal, the Licensee had partially followed the model RETS format (NUREG-0472) for PWRs. In an initial evaluation by the Franklin Reseach Center (FRC), an independent review team, the Licensee's RETS and ODCM submittals were compared with the model RETS (NUREG-0472, Revision 2) and assessed for compliance with the stipulated provisions. Copies of the draft review, dated February 15, 1982 [15, 16], were delivered to the NRC and the Licensee prior to a site visit by the reviewers.

The site visit was conducted on March 11-12, 1982 by the reviewers. Participation from NRC staff was not available. Discussions focused on the initial review of the proposed changes to the RETS and on the technical approaches for an ODCM. The deficiencies in the Licensee's proposed RETS were considered, deviations from NRC guidelines were pointed out, many differences were clarified, and only a few items remained unresolved pending justification by the Licensee. These issues are summarized in Reference 17.

The final version of the Ginna RETS [18], dated August 12, 1982, was submitted to the NRC and transmitted to the FRC reviewers. On January 10, 1983, the reviewers received a draft ODCM [19] from the Licensee. Both documents were subsequently reviewed. The Licensee also made a commitment [20] to correct the deficiencies found in the draft ODCM* submittal. Final evaluation of RETS was detailed in the comparison report [21], which used the draft version of NUREG-0472, Revision 3 [12] as guidance to evaluate the Licensee's submittal. The comparison report also incorporates NRC comments [22, 23], which serve as additional guidelines regarding plant-specific issues.

*It is anticipated that the Licensee's final ODCM submittal will be due shortly after this TER is completed. Thus, the TER includes the evaluation of the Licensee's draft ODCM, in anticipation that all deficiencies will be resolved in the Licensee's final submittal.

2. REVIEW CRITERIA

Review criteria for the RETS and ODCM were provided by the NRC in three documents:

NUREG-0472, RETS for PWRs

NUREG-0473, RETS for BWRs

NUREG-0133, Preparation of RETS for Nuclear Power Plants.

Twelve essential criteria are given for the RETS and ODCM:

1. All significant releases of radioactivity shall be controlled and monitored.

2. Offsite concentrations of radioactivity shall not exceed the 10CFR20, Appendix B, Table II limits.

3. Offsite doses of radioactivity shall be ALARA.

4. Equipment shall be maintained and used to keep offsite doses ALARA.

5. Radwaste tank inventories shall be limited so that failures will not cause offsite doses exceeding 10CFR20 limits.

6. Waste gas concentrations shall be controlled to prevent explosive mixtures.

7. Wastes shall be processed to shipping and burial ground criteria under a documented program, subject to quality assurance verification.

8. An environmental monitoring program, including a land-use census, shall be implemented.

9. The radwaste management program shall be subject to regular audits and reviews.

10. Procedures for control of liquid and gaseous effluents shall be maintained and followed.

11. Periodic and special reports on environmental monitoring and on releases shall be submitted.

12. Offsite dose calculations shall be performed using documented and approved methods consistent with NRC methodology.

Subsequent to the publication of NUREG-0472 and NUREG-0473, the NRC staff issued guidelines [24, 25], clarifications [26, 27], and branch positions [28, 29, 30] establishing a policy that requires the licensees of operating reactors to meet the intent, if not the letter, of the model RETS provisions. The NRC branch positions issued since the RETS implementation review began have clarified the model RETS implementation for operating reactors.

The review of the ODCM was based on the following NRC guidelines: Branch Technical Position, "General Content of the Offsite Dose Calculation Manual" [31]; NUREG-0133 [9]; and Regulatory Guide 1.109 [32]. The ODCM format is left to the licensee and may be simplified by tables and grid printouts.

3. TECHNICAL EVALUATION

3.1 GENERAL DESCRIPTION OF RADIOLOGICAL EFFLUENT SYSTEM

This section briefly describes the liquid and gaseous effluent radwaste treatment systems, release paths, and control systems installed at R. E. Ginna Nuclear Power Plant, a pressurized water reactor (PWR).

3.1.1 Radioactive Liquid Effluent

The liquid radwaste system consists of treatment of the reactor coolant drain tank (normally recycled through the chemical volume control system), steam generator blowdown drains (normally recycled), hot lab drains, equipment and chemical drains, auxiliary building sumps, and intermediate building drains effluents. The effluents are pumped to the waste holdup tank, which is then directed to the waste evaporator for removal of solids. The liquid effluent is then passed through the mixed bed demineralizer to the waste condensate tanks for final discharge to Lake Ontario. This release path constitutes the liquid radwaste effluent line and is monitored by the effluent monitor R-18 (see Figure 1), which provides automatic isolation. For this effluent path, a substream monitor is also installed on the steam generator blowdown drains (monitor R-19).

Radiation monitors are also installed on other effluent lines such as the turbine building floor drain (monitor R-21), the high conductivity waste effluent (monitor R-22), the containment fan cooler (monitor R-16), and the spent fuel pool heat exchanger (monitor R-20). The latter two effluent lines constitute the service water discharge, which also leads to Lake Ontario.

3.1.2 Radioactive Gaseous Effluent

The process gaseous wastes are collected mainly in the chemical and volume control system (CVCS) holdup tank and then compressed to the waste gas decay tanks before being discharged through a charcoal adsorber to the plant vent, as shown in Figure 2. Also discharging to the plant vent is the auxiliary building ventilation system. Monitoring (R-14, R-13, R-10B, or

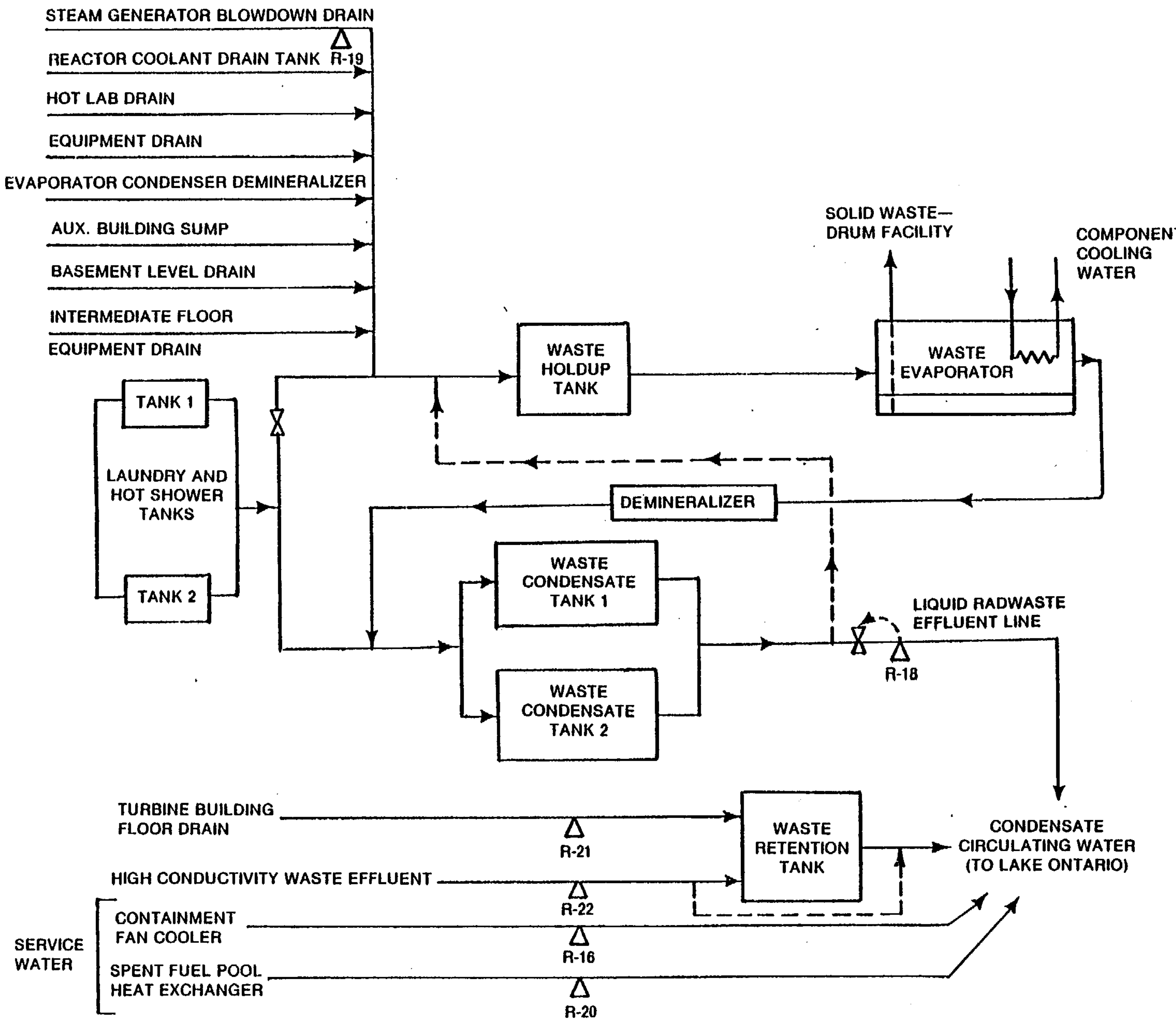

Figure 1. Liquid Radwaste Treatment Systems, Effluent Paths, and Controls for Ginna Nuclear Power Plant

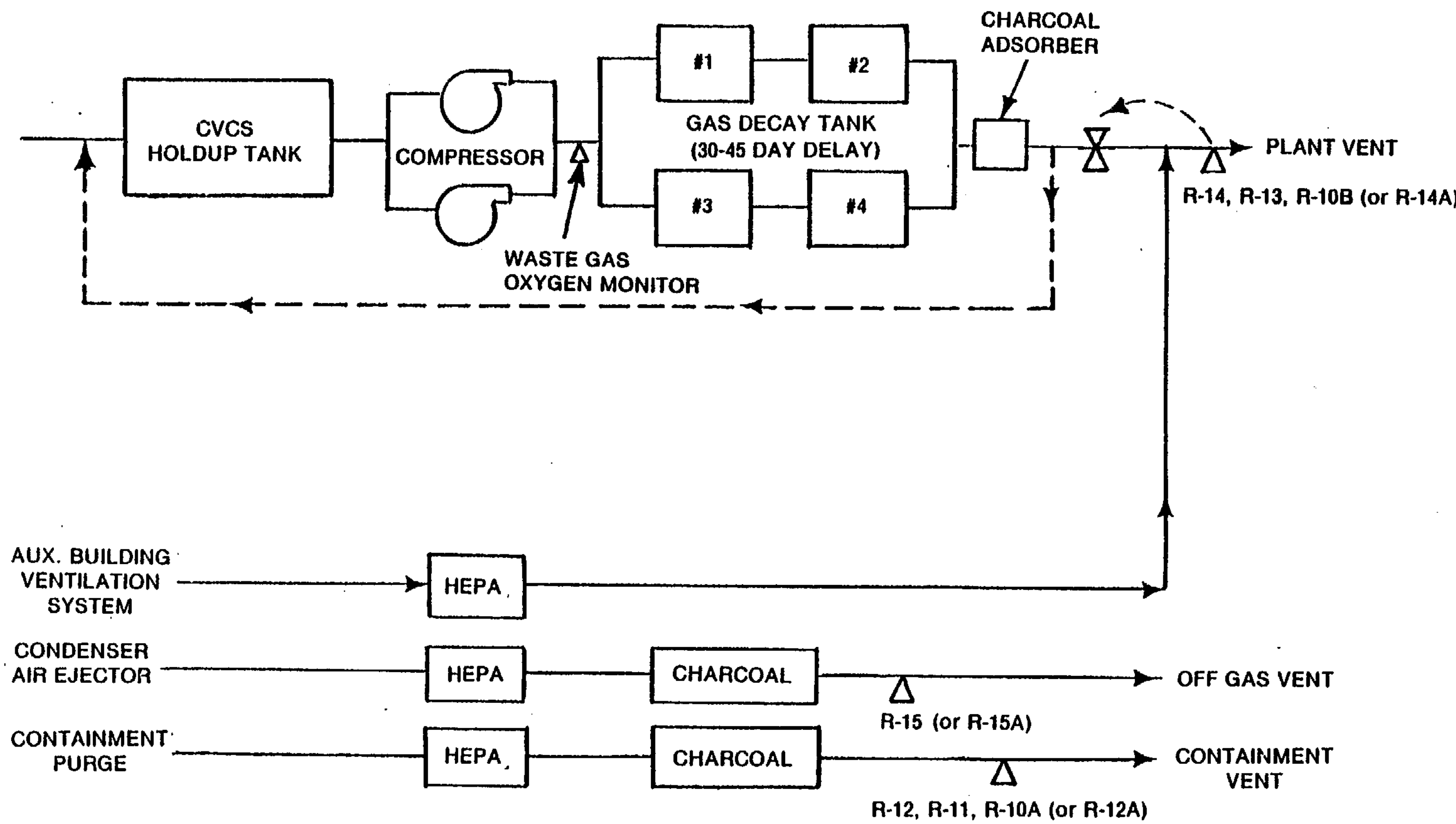

Figure 2. Gaseous Radwaste Treatment Systems, Effluent Paths, and Controls for Ginna Nuclear Power Plant

R-14A) is provided at the plant vent, with noble gas monitor R-14 having the capability to isolate the discharge from the waste decay tanks. The Licensee treats the releases from the plant vent as mixed level releases.

A separate effluent line for the containment purge passes the effluent releases through high-efficiency particulate air (HEPA) filters and charcoal adsorbers to the containment vent, where monitoring (R-12, R-11, R-10A, or R-12A) is provided. Releases from the containment vent are also treated as mixed level. The third effluent line is the offgas vent, which handles the effluents from the condenser air ejector. The effluent line also passes the releases through the HEPA filters and charcoal adsorbers. The effluent line has monitor R-15 or R-15A. Since the offgas vent is located on the roof of the turbine building, its release has been treated as ground level.

3.2 RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS

The evaluation of the Licensee's proposed RETS against the provisions of NUREG-0472 included the following: (1) a review of information provided in the Licensee's 1979 submittal [13, 14], (2) the resolution of problem areas in that submittal by means of a site visit [15, 16, 17], and (3) a review of the Licensee's August 1982 RETS submittal [18] and the January 1983 draft ODCM submittal [19].

3.2.1 Effluent Instrumentation

The objective of the RETS with regard to effluent instrumentation is to ensure that all significant releases of radioactivity are monitored. The RETS specify that all effluent monitors be operable and alarm/trip setpoints be determined to ensure that radioactivity levels do not exceed the maximum permissible concentration (MPC) set by 10CFR20. To further ensure that the instrumentation functions properly, surveillance requirements are needed in the specifications.

3.2.1.1 Radioactive Liquid Effluent Monitoring Instrumentation

The Licensee has provided radiation monitors for potential liquid effluent lines. In addition, automatic isolation is provided for the liquid radwaste effluent line, which is the major effluent release line. It is thus determined that the Licensee's proposal on liquid effluent monitoring instrumentation has satisfied the intent of NUREG-0472 [1, 12].

3.2.1.2 Radioactive Gaseous Effluent Monitoring Instrumentation

The Licensee has provided radiation monitors for potential gaseous effluent lines, for which automatic isolation is also provided for the release from the waste gas decay tanks. It is thus determined that the Licensee's proposal on gaseous effluent monitoring instrumentation has satisfied the intent of NUREG-0472.

3.2.2 <u>Concentration and Dose Rates of Effluents</u>

3.2.2.1 Liquid Effluent Concentration

In Section 3.9.1.1 of the Licensee's submittal, a commitment is made to maintain the concentration of radioactive liquid effluents released from the site to within 10CFR20 limits, and, if the concentration of liquid effluents exceeds these limits, the concentration will be restored as soon as practical to a value equal to or less than the MPC specified in 10CFR20. All batches of radioactive liquid effluents from the release tanks are sampled and analyzed in accordance with a sampling and analysis program which meets the intent of NUREG-0472. Continuous releases are from the waste retention tank and service water effluent discharges from the containment fan cooler and the spent fuel pool heat exchanger. These releases are sampled periodically in accordance with a sampling and analysis program (Table 4.2-1 of the Licensee's submittal), which meets the intent of NUREG-0472.

The liquid radwaste effluent line monitor is provided with alarm and automatic-termination-of-release capability to prevent the release of liquid effluents with a high concentration of radioactive material, which also meets the intent of NUREG-0472.

3.2.2.2 Gaseous Effluent Dose Rate

In Section 3.9.2.1 of the Licensee's submittal, a commitment is made to maintain the offsite dose rate from radioactive gaseous effluents to within

10CFR20 limits, or the equivalent dose rate values prescribed by Section 3.11.2.1 of NUREG-0472. If the dose rate of gaseous effluents exceeds these limits, it will be restored as soon as is practical to a value equal to or less than these limits.

The radioactive gaseous waste sampling and analysis program (Table 4.12-2 of the Licensee's submittal) provides adequate sampling and analysis of the vent discharges, including the substreams, and therefore meets the intent of NUREG-0472.

3.2.3 Offsite Doses from Effluents

The objective of the RETS with regard to offsite doses from effluents is to ensure that offsite doses are kept ALARA and are in accordance with 10CFR50, Appendix I, and 40CFR190. The Licensee has made a commitment to (1) meet the quarterly and yearly dose limitations for liquid effluents, per Section II.A of Appendix I, 10CFR50; (2) restrict the air doses for beta and gamma radiation in unrestricted areas as specified in 10CFR50, Appendix I, Section II.B; (3) maintain the dose level at the site boundary from release of radioiodines, radioactive materials in particulate form, and radionuclides other than noble gases with half lives greater than 8 days within the design objectives of 10CFR50, Appendix I, Section II.C; and (4) limit the annual dose from radioactive materials from the plant at the site boundary to within the requirements of 40CFR190. In each pertinent section, the Licensee has made a commitment to perform dose calculations in accordance with methods given in the ODCM. This satisfies the intent of NUREG-0472.

3.2.4 Effluent Treatment

The objectives of the RETS with regard to effluent treatment are to ensure that wastes are treated to keep releases ALARA and to satisfy the requirement for Technical Specifications governing the maintenance and use of radwaste treatment equipment. The Licensee has made a commitment to use the liquid (Section 3.9.1.3 of the Licensee's submittal) and gaseous (Section 3.9.2.3 of the Licensee's submittal) radwaste treatment systems when the doses averaged over 31 days exceed 25% of the annual dose design objectives, prorated monthly. The Licensee has also made a commitment in the ODCM to calculate the dose monthly. It is determined that the Licensee's proposal meets the intent of 10CFR50, Appendix I, Section II.D.

3.2.5 Tank Inventory Limits

The objective of the RETS with regard to tank inventory limits is to ensure that the rupture of a radwaste tank would not cause offsite doses greater than the limits set in 10CFR20 for non-occupational exposure. The Licensee has not provided a limit for liquid tanks since the Licensee does not intend to use any outside temporary tanks. For gas storage tanks, a limit of 1.0×10^5 curies has been set for noble gases (Section 3.9.2.6 of Licensee's submittal). The Licensee's commitment to comply with tank inventory limits satisfies the intent of NUREG-0472.

3.2.6 Explosive Gas Mixtures

The objective of the RETS with regard to explosive gas mixtures is to prevent hydrogen explosions in waste gas systems. The Licensee has made a commitment (Section 3.9.2.5 of the Licensee's submittal) to maintain a safe concentration of oxygen in the waste gas holdup system by continuous O_2 monitoring, using a minimum of one channel (Table 3.5-6 of the Licensee's submittal) instead of two channels as specified by NUREG-0472. The plant does not have either of the two hydrogen monitors specified in NUREG-0472, Table 3-3.13, Section 2B, for systems not designed to withstand a hydrogen explosion. However, the Licensee treats the system as a hydrogen-rich system. In accordance with the NRC staff position, the present monitoring system is acceptable on an interim basis.

3.2.7 Solid Radwaste System

The objective of the RETS with regard to the solid radwaste system is to ensure that radwaste will be properly processed and packaged before it is shipped to the burial site. Specification 3.11.3 of NUREG-0472 provides for the establishment of a Process Control Program (PCP), or the equivalent, to show compliance with this objective. The Licensee has made a commitment (Section 3.9.2.7 of the Licensee's submittal) to implement such a program in accordance with a PCP and to thus assure that radwaste is properly processed and packaged before it is shipped to the burial site. This meets the intent of NUREG-0472.

3.2.8 Radiological Environmental Monitoring Program

The objectives of the RETS with regard to environmental monitoring are to ensure that an adequate and full-area-coverage environmental monitoring program exists and that the 10CFR50, Appendix I requirements for technical specifications on environmental monitoring are satisfied. In most cases, the Licensee has followed NUREG-0472 guidelines, including the Branch Technical Position dated November 1979, and has provided an adequate number of sample locations for pathways identified. The Licensee's methods of analysis and maintenance of yearly records satisfy the NRC guidelines and meet the intent of 10CFR50, Appendix I. The specification for the land use census satisfies the provisions of Section 3.12.2 of NUREG-0472 by providing for an annual census in the specified areas. The Licensee participates in an interlaboratory comparison program approved by the NRC and reports the results in the Annual Radiological Environmental Operating Report, which also meets the intent of NUREG-0472.

3.2.9 Audits and Reviews

The objective of the RETS with regard to audits and reviews is to ensure that audits and reviews of the radwaste and environmental monitoring programs are properly conducted. The Licensee's administrative structure designates the Plant Operations Review Committee (PORC) and Nuclear Safety Audit and Review Board (NSARB) as the two groups responsible for the review and audit of the radiological environmental monitoring program, the ODCM, and the PCP. The proposed quality assurance (QA) program has met the criteria of 10CFR50, Appendix B. The PORC is responsible for reviewing the procedures associated with these programs. The NSARB is responsible for auditing the program as often as is specified under NUREG-0472.

3.2.10 Procedures and Records

The objective of the RETS with regard to procedures is to satisfy the requirement for written procedures for implementing the ODCM, the PCP, and the QA program. It is also an objective of RETS to properly retain the documented records in relation to the environmental monitoring program and certain QA procedures. The Licensee has made a commitment to establish, implement, and maintain written procedures for the PCP and the ODCM program. The Licensee chooses to maintain the QA program in the existing technical specifications rather than the one specified in the RETS, a practice accepted by the NRC staff. The Licensee intends to retain the records of off-site environmental monitoring surveys and radioactivity environmental releases, as well as records of quality assurance activities for the duration of the facility operating license. It is thus determined that the Licensee has met the intent of NUREG-0472.

3.2.11 Reports

The objective of the RETS with regard to administrative controls is to ensure that appropriate periodic and special reports are submitted to the NRC, and that these reports meet the requirements of 10CFR50.36a.

3.2.11.1 Routine Reports

In Section 6.9.1.3 of the Licensee's submittal, a commitment is made to provide an annual radiological environmental operating report that includes summaries, interpretations, and statistical evaluation of the results of the environmental surveillance program. The report also includes the results of participation in an interlaboratory comparison program specified by Specification 3.12.3 of NUREG-0472 [1,12].

In Section 6.9.1.4 of the Licensee's submittal, a commitment is made to provide semiannual radioactive effluent release reports which include a summary of radioactive liquid and gaseous effluents released, an assessment of offsite doses, and a summary of radioactive solid waste releases. Results of the land use census as well as major changes to radioactive waste treatment systems are also included in the report. These reporting commitments meet the provisions of NUREG-0472.

3.2.11.2 Non-Routine Reports

In the Licensee's submittal, a commitment is made to provide a 30-day written report (according to Section 6.9.2.b of the Licensee's existing technical specifications) for each of the following in NUREG-0472:

- o exceeding liquid effluent dose limits specified in Specifications 3.11.1.2 and 3.11.1.3
- o exceeding gaseous effluent dose rate limits specified in Specifications 3.11.2.2, 3.11.2.3, and 3.11.2.4
- o exceeding total dose limits specified in Specification 3.11.4
- o measured levels of radioactivity in an environmental sampling medium determined to exceed the reporting level of Table 3.12.2.

These reporting commitments have satisfied the provisions of NUREG-0472 [1, 12].

3.2.12 Implementation of Major Programs

One objective of the administrative controls is to ensure that implementation of major programs such as PCP, ODCM, and major changes to the radioactive waste treatment system follow appropriate administrative procedures. The Licensee has made a commitment to review, report, and implement major programs such as PCP, ODCM, and major changes to the radioactive waste treatment system. This commitment meets the intent of NUREG-0472.

3.3 OFFSITE DOSE CALCULATION MANUAL (ODCM)

As specified in NUREG-0472, the ODCM is to be developed by the Licensee to document the methodology and approaches used to calculate offsite doses and maintain the operability of the effluent systems. As a minimum, the ODCM should provide equations and methodology for the following topics:

- o alarm and trip setpoint on effluent instrumentation
- o liquid effluent concentration in unrestricted areas
- o gaseous effluent dose rate at or beyond the site boundary
- o liquid and gaseous effluent dose contributions
- o liquid and gaseous effluent dose projections.

In addition, the ODCM should contain flow diagrams defining the treatment paths and the components of the radioactive liquid, gaseous, and solid waste management systems. Of course, these diagrams should be consistent with the systems being used at the station. A description and location of samples in support of the environmental monitoring program are also needed in the ODCM.

3.3.1 Evaluation

The Licensee has followed the methodology of NUREG-0133 [9] to determine the alarm and trip setpoints for the liquid and gaseous effluent monitors. A conservative factor of 10 is used for the setpoints, which ensures that the maximum permissible concentration (MPC), as specified in 10CFR20, will not be exceeded even in the case of simultaneous discharge from various liquid or gaseous release points.

The Licensee demonstrated the method of calculating the radioactive liquid concentration by describing in the ODCM the means of collecting and analyzing representative samples prior to and after releasing liquid effluents into the circulating water discharge. The method provides added assurance of compliance with 10CFR20 for liquid releases.

Methods are also included for showing that dose rates at or beyond the site due to noble gases, radioiodines, particulates, and radionuclides other than noble gases with half-lives greater than 8 days are in compliance with 10CFR20. In this calculation, the Licensee has considered effluent releases from the plant vent, the containment vent, and the offgas vent; releases from the plant vent and containment vent are treated as mixed level; and releases from the offgas vent are treated as ground level. In all cases, the Licensee has used the highest annual average values of relative concentration (X/Q) and relative deposition (D/Q) to determine the controlling locations. The Licensee intends to use the maximally exposed individual and the critical organ as the reference receptor. The Licensee has also considered pathways from inhalation, food, and ground-plane contaminations, although the ingestion pathways from the ground deposition are not strictly required for gaseous dose rate considerations. The Licensee has demonstrated that the described methods and relevant parameters have followed the conservative approaches provided by NUREG-0133 and Regulatory Guide 1.109.

Evaluation of the cumulative dose is to ensure that the quarterly and annual dose design objectives specified in RETS are not exceeded.

For liquid releases, the Licensee has identified drinking water and fish consumption as the two viable pathways. In the calculation, the Licensee has used nearfield and farfield dilution factors specific to the plant; all other key parameters follow the suggested values given in Regulatory Guide 1.109. As in the case of dose rate calculation, the Licensee has used the maximally exposed individual as the reference receptor. To correctly assess the cumulative dose, the Licensee intends to estimate the dose once per 31 days.

Evaluation of the cumulative dose from noble gas releases includes both beta and gamma and air doses at and beyond the site boundary. The critical organs under consideration are the total body and skin for gamma and beta radiation, respectively. Again, the Licensee has used the maximum (X/Q) values as discussed earlier and has followed the methodology and parameters of NUREG-0133 and Regulatory Guide 1.109.

For radioiodines, particulates, and radionuclides other than noble gases with half-lives greater than 8 days, the Licensee has provided a method to demonstrate that cumulative doses calculated from the release meet both quarterly and annual design objectives. The Licensee has demonstrated a method of calculating the dose using maximum annual average (X/Q) values for the inhalation pathway and has included (D/Q) values for the food and ground-plane pathways, which is consistent with the methodology of NUREG-0133.

Using the existing methodology for gaseous and liquid dose calculations, the Licensee has demonstrated a procedure to determine the monthly dose and to ensure that the design objectives for the liquid radwaste system, the gaseous radwaste system, and the ventilation exhaust system are not exceeded.

Adequate flow diagrams defining the effluent paths and components of the radioactive liquid and gaseous waste treatment systems have been provided by the Licensee. Radiation monitors specified in the Licensee-submitted RETS are also properly identified in the flow diagrams.

The Licensee has provided a description of sampling locations in the ODCM and has identified them in Figures 3 through 6 of that document. This description is consistent with the sampling locations specified in the Licensee's RETS Table 3.16-1 on environmental monitoring.

4. CONCLUSIONS

Table 1 summarizes the results of the final review and evaluation of the R. E. Ginna Nuclear Power Plant proposed Radiological Effluent Technical Specifications (RETS). The review concludes that the Licensee's proposed RETS meets the intent of the NRC staff's current standard, "Radiological Effluent Technical Specifications," NUREG-0472. However, there are minor discrepancies found in the Licensee's submittal; the NRC staff [22, 23] has indicated that corrective changes will be initiated by the NRC project manager so that appropriate wording or information is incorporated into the Licensee's RETS to facilitate the final implementation. These discrepancies are:

1. In table notations (1) and (2) of the Licensee's Table 4.1-5, the Licensee has not addressed automatic pathway isolation and/or control room alarm annunciation under the following conditions: downscale failure, circuit failure, and controls not set in operate mode. The Licensee-provided basis does not adequately clarify the discrepancy.

2. The Licensee has not provided information, equivalent to Figure 5.1-3 of the model RETS [12], containing a site map to clearly define the unrestricted areas within the site boundary with respect to radioactive gaseous and liquid effluent releases.

3. Under the Licensee's Section 6.9.1.4, Semiannual Radioactive Effluent Release Report, the content of the report should be expanded by including the following sentence, "This same report shall include an annual summary of hourly meteorological data collected over the previous year." The sentence can be footnoted so that the Licensee has the option of retaining this summary on site in a file that shall be provided to the NRC upon request.

4. The Licensee should make a commitment in the administrative control sections that the Licensee-initiated changes to ODCM (Section 6.15), PCP (Section 6.15), and major changes to the radioactive waste treatment system (Section 6.17) shall become effective upon review and acceptance by the PORC.

The review also concludes that the Licensee's Offsite Dose Calculation Manual (ODCM) uses documented and approved methods that are consistent with the criteria of NUREG-0133.

Table 1. Evaluation of Proposed Radiological Effluent Technical Specifications (RETS), Ginna Nuclear Power Plant

RETS Requirement	Technical Specifications: NRC Staff Model RETS NUREG-0472 (Section)*	Technical Specifications: Licensee Proposal (Section)	Replaces or Updates Existing Tech. Spec. (Section)	Evaluation
Effluent Instrumentation	3/4.3.3.10, 3/4.3.3.11	3.5, 4.0	3.5, 4.0	Meets the intent of NRC criteria
Radioactive Effluent Concentrations	3/4.11.1.1, 3/4.11.2.1	3.9.1.1, 4.12.1.1 3.9.2.1, 4.12.2.1	3.9.1, 4.12.1.1 3.9.2, 4.12.2.1	Meets the intent of NRC criteria
Offsite Doses	3/4.11.1.2, 3/4.11.2.2, 3/4.11.2.3, 3/4.11.4	3.9.1.2, 4.12 3.9.2.2, 3.9.2.4	Not addressed	Meets the intent of NRC criteria
Effluent Treatment	3/4.11.1.3, 3/4.11.2.4	3.9.1.3, 4.0 3.9.2.3	Not addressed	Meets the intent of NRC criteria
Tank Inventory Limits	3/4.11.1.4, 3/4.11.2.6	3.9.2.6, 4.12.3	Not addressed	Meets the intent of NRC criteria
Explosive Gas Mixtures	3/4.11.2.5	3.9.2.5	Not addressed	Meets the intent of NRC criteria in the interim
Solid Radioactive Waste	3/4.11.3	3.9.2.7	Not addressed	Meets the intent of NRC criteria
Environmental Monitoring	3/4.12.1	3.1.6, 4.10.1	Not addressed	Meets the intent of NRC criteria
Audits and Reviews	6.5.1, 6.5.2	6.5.1, 6.5.2	6.5.1, 6.5.2	Meets the intent of NRC criteria
Procedures and Records	6.8, 6.10	6.8, 6.10	6.8, 6.10	Meets the intent of NRC criteria
Reports	6.9.1.11, 6.9.1.12, 6.9.2, 6.10.2,	6.9.1.2, 6.9.1.3, 6.9.1.4, 6.5.2.10,	6.9.3.a, 6.9.3.b, 6.9.2, 6.10.2	Meets the intent of NRC criteria
Implementation of Major Programs	6.13, 6.14, 6.15	6.15, 6.16, 6.17	Not addressed	Meets the intent of NRC criteria

*Section numbering sequence is according to NUREG-0472, Rev. 3 [12].

5. REFERENCES

1. "Radiological Effluent Technical Specifications for Pressurized Water Reactors," Rev. 2
   NRC, July 1979
   NUREG-0472

2. Title 10, Code of Federal Regulations, Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion, 'As Low As Is Reasonably Achievable,' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents"

3. Title 10, Code of Federal Regulations, Part 50, Appendix I, Section V, "Effective Dates"

4. Title 10, Code of Federal Regulations, Part 20, "Standards for Protection Against Radiation"

5. Title 40, Code of Federal Regulations, Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operations"

6. Title 10, Code of Federal Regulations, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants"

7. Title 10, Code of Federal Regulations, Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"

8. "Radiological Effluent Technical Specifications for Boiling Water Reactors," Rev. 2
   NRC, July 1979
   NUREG-0473

9. "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants, A Guidance Manual for Users of Standard Technical Specifications"
   NRC, October 1978
   NUREG-0133

10. C. Willis and F. Congel (NRC)
    "Summary of Draft Contractor Guidance of RETS"
    Presented at the AIF Environmental Subcommittee Meeting, Washington, DC
    May 19, 1982

11. F. Congel (NRC)
    Memo to RAB Staff (NRC)
    Subject: Interim Changes in the Model Radiological Effluent Technical Specifications (RETS)
    August 9, 1982

12. "Radiological Effluent Technical Specifications for Pressurized Water Reactors," Rev. 3, Draft 7, intended for contractor guidance in reviewing RETS proposal for operating reactors
NRC, September 1982
NUREG-0472

13. Ginna Radiological Effluent Technical Specifications
Rochester Gas and Electric Company, February 14, 1979
NRC Docket No. 50-244

14. Ginna Offsite Dose Calculation Manual
Rochester Gas and Electric Company, February 14, 1979
NRC Docket No. 50-244

15. "Comparison of Specification NUREG-0472, Radiological Effluent Technical Specifications for PWRs, vs. Licensee Submittal of Radiological Effluent Technical Specifications for Ginna Nuclear Power Plant" (Draft)
Franklin Research Center, February 15, 1982

16. Technical Review of Offsite Dose Calculation Manual for Ginna Nuclear Power Plant (Draft)
Franklin Research Center, February 15, 1982

17. Franklin Research Center
Letter of Transmittal to NRC
Subject: Trip report on site visit to Ginna Nuclear Power Plant
March 26, 1982

18. Radiological Effluent Technical Specifications (RETS)
Ginna Nuclear Power Plant
August 12, 1982
NRC Docket No. 50-244

19. Ginna Offsite Dose Calculation Manual (Draft)
Rochester Gas and Electric Company, January 1983
NRC Docket No. 50-244

20. S. Pandey/S. Chen (FRC)
Memo to W. Meinke (NRC)
Subject: Telephone Communication with the Licensee (Ginna) on Draft ODCM
January 12, 1983

21. "Comparison of Specification NUREG-0472, Radiological Effluent Technical Specifications for PWRs, vs. Licensee Submittal of Radiological Effluent Technical Specifications for Ginna Nuclear Power Plant"
Franklin Research Center, December 15, 1982

22. W. Meinke (NRC)
Memo to A. Cassell (FRC)
Subject: Comments to FRC RETS Review on Ginna Power Plant
November 10, 1982

23. W. Meinke (NRC)
Memo to S. Pandey (FRC)
Subject: Additional Comments to FRC RETS Review on Ginna Power Plant
December 9, 1982

24. C. Willis (NRC)
Letter to Dr. S. Pandey (FRC)
Subject: Changes to RETS requirements following meeting with Atomic Industrial Forum (AIF)
November 20, 1981

25. C. Willis (NRC)
Letter to Dr. S. Pandey (FRC)
Subject: Control of explosive gas mixture in PWRs
December 18, 1981

26. C. Willis and F. Congel (NRC)
"Status of NRC Radiological Effluent Technical Specification Activities"
Presented at the AIF Conference on NEPA and Nuclear Regulations, Washington, D.C.
October 4-7, 1981

27. C. Willis (NRC)
Memo to P. C. Wagner (NRC)
"Plan for Implementation of RETS for Operating Reactors"
November 4, 1981

28. W. P. Gammill (NRC)
Memo to P. C. Wagner (NRC)
"Current Position on Radiological Effluent Technical Specifications (RETS) Including Explosive Gas Controls"
October 7, 1981

29. "An Acceptable Radiological Environmental Monitoring Program"
Branch Technical Position
November 1979

30. Methods for Demonstrating LWR Compliance with the EPA Uranium Fuel Cycle Standard (40CFR190)
NRC, February 1980
NUREG-0543

31. "General Contents of the Offsite Dose Calculation Manual," Revision 1
Branch Technical Position, Radiological Assessment Branch
February 8, 1979

32. Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR50, Appendix I
NRC, October 1977
Regulatory Guide 1.109